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Collaborative Innovation in Radioactive Waste Management Technology
: Industry, Academia, and Research Together



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1분과

핵주기정책·규제 및 비확산 (ORAL)

Development and Application of Safeguardability Evaluation Method (SEM) to Improve Safeguards by Design (SBD)

Jiyoung Kim and Seung Min Woo*

Kyung Hee University

*woosm@khu.ac.kr

As the development of advanced nuclear facilities progresses rapidly, the importance of Safeguards by Design (SBD), which integrates safeguards at the design stage, is increasing. Implementing SBD can lead to robust safeguards, cost efficiency, and enhanced proliferation resistance. Safeguardability, which measures the ease of safeguards, is a concept that can be applied within the SBD framework. Two representative methods for evaluating safeguardability are the Facility Safeguardability Analysis (FSA) and the Safeguardability Check-List (SCL). However, these methods have limitations: FSA is not quantitative, and the evaluation criteria in SCL lack clarity. To address these issues, this study developed a new evaluation method called the Safeguardability Evaluation Method (SEM), which incorporates evaluation criteria based on formulas.

SEM consists of three categories such as Nuclear Material Accountancy (NMA), Containment & Surveillance (C&S), and Design Information Verification (DIV). For example, NMA requires the Material Balance Area (MBA) of the target facility and performs Material Balance Evaluation (MBE) based on quantitative evaluation of nuclear materials entering and leaving the MBA. One of the NMA factors is Material Unaccounted For (MUF). MUF assessment is essential to ensure robust safeguards. MUF is the difference between Book Inventory and Physical Inventory. MUF occurs in situations such as errors occurring in facility processes, measurement errors, or intentional diversion of nuclear materials. MUF must meet IAEA safeguards requirements that are $MUF < 1SQ$ and $3\sigma_{MUF} < 1SQ$. Considering this, σ_{MUF} factors were developed as follows. ($SEM_3 = 1 - \frac{3\sigma_{MUF}}{SQ}$) In this way, 12 factors were developed.

In addition to that, we modeled a virtual nuclear facility and apply SEM developed to conduct a safeguardability evaluation. The virtual nuclear facility is a light water nuclear fuel manufacturing facility with an annual production capacity of 500 MTU, assuming MBA and KMP, and the nuclear material measurement method and uncertainty of the facility were set by referring to the ITV (International Target Value) report for NMA. In addition, modeling was performed for application of C&S factors and SEM was applied to the modeled virtual nuclear facility. Finally, SEM is expected to contribute to improving the design from the perspective of Safeguards by Design (SBD).

Acknowledgements

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Keywords: Safeguardability, Non-proliferation, Safeguards-By-Design, Safeguards

Modeling and Safeguardability Assessment of a Simulated Nuclear Facility Based on i-SMR

Hyunho Kang¹, Seungmin Lee², and Seung Min Woo^{1,*}

¹Kyung Hee University

²Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*woosm@khu.ac.kr

The IAEA recommends implementing the Safeguards-by-Design (SBD) to ensure the effective and efficient application of safeguards. This involves assessing the safeguardability during the design phase of new nuclear facilities and developing strategies to strengthen safeguards. Safeguardability is defined as “the degree of ease with which a nuclear energy system can be effectively and efficiently placed under international safeguards”. However, South Korea currently lacks a regulatory framework for the SBD, and several relevant studies have been conducted to establish such a system.

As part of this effort, the Korea Atomic Energy Research Institute (KAERI) has developed a Safeguardability Evaluation Tool (SET). This tool generates diversion paths using a combination of fault tree and event tree analyses, and quantitatively and qualitatively assesses safeguardability for the nuclear system. The quantitative assessment is based on the detection probability, the diversion time, and the detection efficiency, while the qualitative assessment uses a checklist to evaluate the ease of nuclear material accountancy, containment and surveillance, and design information verification. To verify the applicability of the SBD, a simulated new nuclear facility based on the Innovative Small Modular Reactor (i-SMR) under development in South Korea was modeled and its safeguardability was assessed with a focus on the detection probability.

The i-SMR system is currently under development that is why there is the limitation and lack of detail information for the system. Therefore, a simulated new nuclear facility was modeled using four modules based on the latest design, which includes a spent fuel storage pool at the end of the module. The model utilized information on Westinghouse 17×17 fuel, similar to the fuel used in the System-integrated Modular Advanced Reactor (SMART) system, as well as the fuel building and the fuel handling systems from APR1400. The safeguard system was established following the IAEA’s light-water reactor safeguard approaches.

As a result of this study, it was found that the number of diversion paths decreased compared to the previous design, where the spent fuel storage pool was in the center and two reactor modules were on each side. This reduction of diversion paths is due to the decrease in the number of fuel transfer tube. Additionally, it was identified that the greatest number of diversion paths is generated in the Fuel Loading/Unloading Bay, Fuel Transfer Tube, and Spent Fuel Storage Pool, making them more vulnerable to diversion, with spent fuel being more vulnerable to diversion than fresh fuel.

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Keywords: Safeguards-by-Design (SBD), Safeguardability, Diversion paths, Innovative Small Modular Reactor (i-SMR), Safeguardability Evaluation Tool (SET)

Establishment of Spent Fuel Composition Database for SMR Safeguards Purposes and Its Application for the Non-proliferation Perspective

Jeemin Han and Seung Min Woo*

Kyung Hee University

*woosm@khu.ac.kr

The Innovative Small Modular Reactor (i-SMR), currently under development in South Korea, is expected to have different safeguards compared to conventional reactors due to its innovative features such as the modular design, the load-following operation and the spent fuel management. One of the crucial information in terms of safeguards and non-proliferation is the spent fuel composition data. Therefore, in this study, the database of spent fuel composition for i-SMR was developed. Since detailed design information for i-SMR has not yet been developed, there are limitations in modeling the reactor system. Therefore, the System-Integrated Modular Advanced Reactor (SMART), developed by the Korea Atomic Energy Research Institute and approved for standard design, is utilized to develop the database system.

In order to achieve the spent fuel composition data, core modeling and burnup simulations were conducted using the Monte Carlo N-Particle (MCNP) 6.2 code. The amount of nuclear material in spent nuclear fuel varies depending on three key factors: initial enrichment, cooling time, and burnup. In this study, we established the spent nuclear fuel database by varying the initial enrichment and cooling time. Additionally, we prepared a case of database for using HALEU by increasing initial enrichment, and we established the database to cover a cooling time of up to 100 years. We created separate sheets for each enrichment level, and we reflected modular operation by designing individual data sheets for each module. Moreover, a single sheet form that includes composition data for all cases was developed. The database format includes not only composition data but also SUM data (total mass) to visualize the total quantity of isotopes. We added core information and organized the sheets by enrichment level to enhance user convenience. Additionally, the Pu Quality value based on composition was also analyzed depending on the simulation case.

As part of the database utilization, nuclear proliferation assessments were conducted using the Figure of Merit (FOM) methodology, which evaluates the attractiveness of plutonium in spent nuclear fuel. As a second step in the non-proliferation study, nuclear proliferation assessments based on the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) methodology will be conducted, and the results of the two methodologies will be compared and discussed.

This study serves as a preliminary research phase for the construction of a spent nuclear fuel material database for i-SMR. From the perspective of nuclear non-proliferation and safeguards, the database format developed in this study will be used to create a material database for i-SMR once detailed design information becomes available.

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Keywords: Spent nuclear fuel, SMR, Safeguards, Figure-of-Merit (FOM), Non-proliferation

Development of the Uranium Verification System for Domestic Environments

Jihyun Ahn¹, Jinho Ryu², Seungho Ahn², Heekyun Baek³, Jungki Shin³, and Hee Seo^{1,*}

¹Jeonbuk National University (JNBU)

²Korea Institute of Nuclear Nonproliferation and Control (KINAC)

³Neosiskorea

*hseo@jbnu.ac.kr

The international community, to ensure the peaceful use of nuclear energy, has established safeguards agreements with the International Atomic Energy Agency. These agreements enable the monitoring and verification of nuclear materials. Among these procedures, uranium verification plays a critical role in confirming the peaceful use of nuclear materials, as it provides accurate information regarding the location and status of such materials. Non-destructive analysis is one of the most widely used methods for uranium verification, offering the advantage of rapid on-site analysis without damaging the materials. Techniques such as gamma spectrometry, a Non-destructive analysis method, are highly useful for measuring uranium enrichment levels.

However, the accuracy of Non-destructive analysis based uranium verification can vary depending on the performance of detectors, analytical conditions, and the characteristics of the samples. Therefore, developing a uranium enrichment analysis system that considers environmental conditions of verification is necessary for improving the accuracy of results. Additionally, as analytical results can be influenced by the user's level of expertise, the development of user-friendly systems that minimize errors and enhance analytical capabilities is required.

This study focuses on establishing a user-friendly uranium verification system optimized for domestic environments, based on the IAEA's uranium verification methodology. The goal is to improve the accuracy and reliability of the domestic verification process. To achieve this, the study analyzes the IAEA's uranium verification methods and examines the environmental conditions for domestic verification targets, such as pellets, fuel rods, and UF₆ cylinders. Based on this analysis, the study proposes an optimal uranium verification method tailored to domestic conditions. Future research will aim to develop a verification system suited to the domestic environment, evaluate its performance through field experiments, and contribute to the establishment of a more efficient uranium verification system. This would not only enhance safety in the management of nuclear materials domestically but also strengthen international confidence.

Acknowledgements

This work was supported by the Korea Institute of Nuclear Nonproliferation and Control (KINAC).

Keywords: Safeguards, Non-destructive assay, Gamma spectroscopy, Nuclear material

Analysis of Proliferation Concerns and Safeguards Technology Development Needs for Molten Salt Reactors

Seong-Kyu Ahn*, Young-Eun Jung, and Jae Soo Ryu

Korea Atomic Energy Research Institute (KAERI)

*skahn76@kaeri.re.kr

Molten salt reactors (MSRs) have experienced a resurgence of interest worldwide over the past decade. In Korea, research is also underway to develop small MSRs for ship propulsion. MSRs come in a variety of designs, ranging from solid fuel to liquid fuel designs. A salt process also can be different (on-line process or off-line process), and the choice of nuclear fuel as well as base salt material varies. As a result, the applying safeguards and the required technologies are much more variable depending on the reactor design than for other types of advanced reactors. As innovative MSR research is rapidly being introduced and evolving, it is necessary to investigate the concerns from a proliferation perspective and draw the direction for the development of corresponding safeguards scheme and necessary technologies. This study analyzed the characteristics of nuclear nonproliferation perspectives and the current status of safeguards technologies for MSRs, focusing on the MSR reactor design concepts currently under development in Korea. In particular, based on a review of the domestic technology level accumulated through the safeguards R&D for pyroprocessing, the study identified ways to utilize them and the technologies and development directions that need to be further developed for effective and efficient safeguards. These results are expected to be utilized in the selection of technology development items to be included in the demonstration research and development of advanced reactors to be promoted, as well as in research planning for the development of nuclear nonproliferation regulatory guidelines.

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Keywords: MSR, Safeguards, Nonproliferation, Nuclear fuel cycle, Strategy

The MACSTOR Lawsuit and the High-Level Radioactive Waste Act

Sangcheol Hyung*, Dongjin Kim, and Hana Seo

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*wakendragon@kinac.re.kr

The Nuclear Safety and Security Commission granted a license to change the operation of the Wolseong Spent Nuclear Fuel Phase II Compact Dry Storage Facility (MACSTOR). In response, local residents filed a claim for invalidation in court.

The lawsuit challenged whether MACSTOR is a “related facility” as a “facility for handling and storing nuclear fuel materials”, and if so, whether an accident management plan should be reviewed, and whether regulations on interim storage facilities for spent nuclear fuel or storage facilities for high-level radioactive waste should be applied, whether the public consultation process was required, and whether MACSTOR was not a facility related to spent nuclear fuel, which is prohibited from being built in an area where a low- and intermediate-level waste disposal facility is located under Article 19 of the Waste Disposal Site Act.

The court held that 1. Spent fuel is nuclear fuel material unless a decision to dispose of it has been made, and dry storage facilities are related facilities because they perform indispensable functions as facilities related to the safety of nuclear reactors. 2. Permission to change operations can be granted without reviewing the accident management plan. 3. The analogy between dry storage facilities and spent fuel interim storage facilities cannot be applied because they are separate and distinct facilities. 4. Related facilities under the Waste Disposal Site Act are facilities of the same or higher level as spent fuel interim storage facilities. 5. the public consultation process is not applicable to this case because it applies to the continued operation of a nuclear power plant after the expiration of its design life.

The proposed High-Level Radioactive Waste Act reflects the issues in the above litigation. KINAC is studying ways to apply safeguards and physical protection for spent nuclear fuel interim storage facilities and high-level radioactive waste disposal facilities.

Keywords: Spent nuclear fuel, Dry storage facility, High-level radioactive waste

1분과

핵주기정책·규제 및 비확산 (POSTER)

Feasibility Study on Using Quasi Reference Electrodes for Monitoring Electric Signals Acquired During Electrodeposition

Young-Eun Jung* and Jonghui Han

Korea Atomic Energy Research Institute (KAERI)

*jungyoungeun@kaeri.re.kr

Pyroprocessing is a technology for managing used nuclear fuels. For a back-end fuel cycle technology, its safeguardsability plays an important role in achieving international credibility and nuclear transparency. The international atomic energy agency (IAEA) recommends a use of supplementary safeguards techniques along with traditional techniques, especially for advanced nuclear facilities, such as pyroprocessing. Process monitoring (PM) is a representative supplementary method. Since it monitors operating states of the process in-situ, it is able to indirectly obtain safeguards-related information in real time. As an effort to enhance safeguardsability of pyroprocessing, it has been investigated to apply PM to pyroprocessing. Electrorefining is a key unit process in pyroprocessing where uranium is selectively separated from used fuels. During the process, several process signals are to be available while uranium ions electrodeposit onto a cathode as a form of uranium metal. One candidate for the signals to be monitored is an electrochemical signal (either cathodic potential or current).

For a study in electrochemistry, a three-electrode system is often employed. It consists of a working electrode, a counter electrode, and a reference electrode (RE). RE constantly maintains its electric potential difference between the interface of electrode and electrolyte as its electrochemical equilibrium rate is fast enough for both oxidation and reduction reactions. As a result, the electric potential of RE is not easily changed when cell potential is shifted. Therefore, RE allows to measure the electric potential of a working electrode.

One representative RE is using $\text{Ag}|\text{Ag}^+$ reaction. In molten salt experiments, which have a high temperature environment, a $\text{Ag}|\text{Ag}^+$ RE consists of Ag wire and AgCl in LiCl-KCl eutectic salt. As the $\text{Ag}|\text{Ag}^+$ RE uses a fragile material for a container to have permeability, such as mullite tube, its durability is not robust. And also, in the long term, the salt in the container is evaporated. A quasi-RE (Q-RE) is a simple metal rod, which can be used as an alternative. Although the half-cell reaction at a Q-RE is not fully characterized thermodynamically, the stability and durability is prevalent.

In this study, we investigate the feasibility of using a Q-RE as an alternative to actual RE. We select tungsten and molybdenum as Q-REs, and test their reproducibility and stability by employing electrochemical techniques. Lastly, we acquire electric potential and current data by electrodeposition experiments and compare the results with those obtained using actual RE.

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Keywords: Pyroprocessing, Safeguards, Electrorefining, Process monitoring, Quasi-reference electrode

Isotopic Ratio Analysis of Uranium Particles in Environmental Samples Using Large Geometry-Secondary Ion Mass Spectrometry (LG-SIMS)

Hyunju Kim^{1,*}, Jeongmin Kim², Yuyoung Lee², Sohee Cha¹, Jung Youn Choi¹, and Haneol Lee¹

¹Korea Institute of Nuclear Nonproliferation and Control (KINAC)

²Korea Basic Science Institute (KBSI)

*hyunju1157@kinac.re.kr

The analysis of uranium (U) isotopes is crucial for verifying nuclear activities and ensuring the absence of undeclared nuclear material at facilities, particularly within the framework of nuclear safeguards-related applications. Environmental samples, typically collected through surface wipe sampling with cotton swipes, represent a widely accepted verification method by the International Atomic Energy Agency (IAEA) for nuclear safeguards.

Until now, the Korea Institute of Nuclear Nonproliferation and Control (KINAC) has relied on external institutions for precise analysis and evaluation of analysis results. However, as domestic nuclear facilities diversify, there is a growing need to expand the types of nuclear materials subject to verification. Additionally, the potential for unexpected results in environmental sample analyses necessitates a robust response. To address these challenges and maintain regulatory independence, it is increasingly important for KINAC to develop its own particle analysis capabilities and enhance analytical technologies for the timely and accurate detection of trace amounts of nuclear materials. Advanced mass spectrometry techniques are essential for the particle analysis of these environmental samples, with analyses currently performed by (Large Geometry-) Secondary Ion Mass Spectrometry ((LG-)SIMS) or Fission Track-Thermal Ionization Mass Spectrometry (FT-TIMS).

To achieve this, we aimed to establish our own analytical capabilities for U particle in environmental samples collected from domestic nuclear facilities. We sought to develop an optimal analytical protocol-covering prescreening, sample preparation, analysis, and evaluation- through the collaborative use of the LG-SIMS instrument (IMS-1300HR³, CAMECA) at the Korea Basic Science Institute (KBSI) Ochang Center.

Prescreening of collected samples for the presence of U particle is conducted by Monochromatic Micro-focusing X-ray Fluorescence (MMXRF) to identify potential candidates for further precise analysis. Selected samples are then processed to recover U particle onto a carbon planchet using vacuum impactor in a clean room, and these planchet samples are analyzed using LG-SIMS to determine the isotopic ratio of the U particles.

We assessed the CRM and actual samples by analyzing U particles and evaluating the results. To improve the precision and accuracy of U particle analysis, we aim to develop optimal analytical methods. This will enhance regulatory agencies' technical capabilities for U particle analysis.

Keywords: Particle analysis, Uranium, Isotopic, LG-SIMS, Nuclear safeguards

Development of a GUI-Based Program for Book Examination and Short Notice Random Inspection to Support National Safeguards Activities

Byung Hee Won*, Hyun-Jo Kim, and Ju-Ang Jung

Korea Atomic Energy Research Institute (KAERI)

*wonbh@kaeri.re.kr

The COISP (Common Inspection On-site Software Package) supports IAEA inspection activities at nuclear facilities by reviewing complex inventory data and identifying items that require further verification. To enhance its national inspection capabilities, Korea is developing a program to assist with national safeguards inspections, incorporating key functions similar to those of CIOSP.

This study focuses on implementing two of CIOSP's primary functions in a similar manner: Book Examination and Short Notice Random Inspections (SNRI). Since the detailed coding of CIOSP is not publicly available, the program developed in this research was designed by analyzing how CIOSP processes sample data and produces results. Additionally, a graphical user interface (GUI) was integrated into the system to ensure it can be used effectively in national inspections.

First, the book examination function was closely analyzed, and an algorithm was developed to produce identical outcomes. This function classifies inventory change list data, provided by facility operators, according to pre-established rules and determines whether the values represent inputs or outputs. It then combines this data with beginning inventory records to calculate the current inventory (ending inventory) amounts for inspectors. Using the developed algorithm, 19 inventory change datasets were processed, and the results matched those produced by CIOSP perfectly. Furthermore, an error-checking algorithm was developed to identify any flaws in each dataset entry prior to the book examination. This algorithm detects errors and pinpoints their exact locations in the data. Comparative tests showed that the error-checking algorithm developed in this study provides more detailed error detection than the CIOSP system.

The second key feature replicates the support provided for SNRI. During SNRI, since the facility is in operation, the inventory verification is not conducted on items currently in the process, but rather on the items that were in inventory before being processed and the final output inventory items. This function estimates the inventory status at the input and output points based on the inventory declaration data submitted by facility operators through the mailbox system. In CIOSP, 90,900 reported mailbox data resulted in the generation of 4,258 final inventory items. The algorithm developed in this study, using the same reported data, produced 4,274 final inventory items. This result is highly consistent with CIOSP's output, with slight differences likely attributable to version discrepancies in CIOSP. A detailed analysis will be conducted in future work to address these differences.

The functionalities developed in this study by benchmarking CIOSP are expected to be incorporated into national inspection support programs and utilized to enhance Korea's safeguards inspection capabilities.

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Keywords: Common inspection on-site software package, Book examination, Short notice random inspections, National safeguards inspection

Deriving a CNN-Based Detection Algorithm Approach for Enhanced Detection of Nuclear Material Loss in NRTA Systems

Byung Hee Won

Korea Atomic Energy Research Institute (KAERI)

wonbh@kaeri.re.kr

Material Balance Evaluation (MBE) is a critical element of safeguards, particularly in facilities that manage significant amounts of plutonium, which can be used to manufacture nuclear weapons even in small quantities. Implementing an NRTA (Near Real Time Accounting) system is essential in such cases. The NRTA system performs material balance evaluations throughout the year with multiple consecutive assessments. This approach enhances the system's timeliness and detection capability for abrupt nuclear material loss.

In NRTA systems, various loss detection algorithms are employed to determine potential nuclear material loss based on continuous material balance values, known as MUF (Material Unaccounted For), and the corresponding covariance matrix. Loss detection algorithms generally work by converting the observed MUF into SITMUF (Standardized Independent Transformed MUF) and then directly or indirectly comparing the SITMUF values with predefined threshold values to determine whether a nuclear material loss has occurred. In this study, we developed an algorithm capable of determining nuclear material loss by applying a deep learning-based method, specifically using Convolutional Neural Network (CNN) model, to analyze observed data. Simulations were conducted to generate MUF, MUF Residual, SITMUF values, and their corresponding covariance matrices, which can be observed and calculated during the operation of a real NRTA system operation. These simulated values were structured into input data with a matrix size of $10 \times 10 \times 3$, which were then used to train the CNN model to assess the potential for nuclear material loss.

The simulation models were designed to target the pyroprocessing facility, assuming that nine material balance evaluations are conducted annually. For training data, between 100,000 and 1 million sets of input data were generated under two conditions: normal operation and a scenario where 8 kg of nuclear material is lost during a single material balance period. Transfer learning was then applied using ResNet, a pre-trained CNN model originally developed for image classification. Performance comparison results showed that the deep learning-based detection algorithm developed in this study outperformed the traditional detection algorithms, GEMUF (geschätzter, or estimated, MUF) test and Shewhart test, by 17.72% and 30.29%, respectively. The technology developed in this study can be actively considered in facilities where improving detection algorithm performance is crucial, especially when performance improvements in other factors are difficult to achieve. However, to apply the deep learning-based detection algorithm to a real NRTA system, further research that considers real-world scenarios will be necessary.

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Keywords: Near real time accounting, Convolutional neural networks, Material balance evaluation, Nuclear material loss detection

Real – Time Monitoring of Strontium in Contaminated Molten Salt Systems by Laser Induced Breakdown Spectroscopy

Junwon Kim, Jihye Park, Hyeongbin Kim, Wonseok Yang, and Sungyeol Choi*

Seoul National University (SNU)

*choisys7@snu.ac.kr

Molten Salt Reactors (MSRs), a next-generation reactor type, are gaining attention for their safety and economic advantages. For stable operation, real-time monitoring of fission products such as Sr-90, which emit high levels of heat and radiation, is crucial. Laser-Induced Breakdown Spectroscopy (LIBS) emerges as a powerful tool for this task. LIBS analyzes the plasma generated by high-pulse lasers targeting materials, allowing for the detection of elemental composition. Its key advantage lies in the fact that it requires no complex sample preparation, making it highly suitable for real-time, online monitoring. In LIBS analysis, as the number of elements in the salt increases, accurately determining the target concentration becomes more difficult due to matrix effects. A significant challenge arises from the presence of insoluble solid impurities in the molten salt. These solid particles can scatter both the laser beam and plasma emissions, introducing matrix effects that distort the LIBS signal, reducing the accuracy of element concentration measurements. Addressing this issue is crucial for reliable real-time monitoring in environments where such impurities are present.

This study aims to demonstrate that the concentration of target elements can be accurately determined in real time using LIBS, even in a complex environment where the molten salt contains multiple elements. The experiment involved dissolving a specific amount of Sr in a molten salt based on a 50 g LiCl-KCl eutectic mixture. By introducing carbon as an impurity into this molten salt, we attempted a correction method to allow real-time Sr measurement via LIBS. The results show that the Sr concentration can be estimated in real time with sufficient accuracy, even in a visually opaque, carbon-contaminated molten salt environment. The estimates closely matched the Sr concentration in carbon-free molten salt, confirming the feasibility of using LIBS for real-time monitoring of target concentrations in a multi-element molten salt environment.

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Keywords: Laser-induced breakdown spectroscopy, Molten salt reactor, Fission products, Real-time detection, Concentration measurement

Analysis of Molten Salt Reactor Safeguards Factors Through Full Core MSRE Simulation

Jihoon Lee and Seung Min Woo*

Kyung Hee University

*woosm@khu.ac.kr

Seaborg Technology, TerraPower, and various governments demonstrated strong interest in advancing molten salt reactors (MSRs), particularly due to their potential for safe and portable energy supplementation. Seaborg Technology, for example, planned to develop protocols by 2027 and aimed to achieve regulatory approval for commercial MSR deployment by 2030. As such, safeguards from the initial design phase became increasingly critical to ensure that MSRs met international safeguards standards and performed reliably over time. This was referred to as Safeguards-By-Design (SBD). The International Atomic Energy Agency (IAEA) recommended applying the SBD concept to design robust safeguards systems and enhance the proliferation resistance of a nuclear system. The analysis of the MSR system was carried out as the first step of SBD.

In this research, a simulation of the Molten Salt Reactor Experiment (MSRE), a research reactor at the Oak Ridge National Laboratory (ORNL), was conducted using the Serpent 2 code to depict its actual four-year operational period. The simulation accounted for real-time operational conditions, including the removal of fission products, mainly noble gases such as Xenon and Krypton, and the refueling of U-233. The depletion simulation consisted of three phases as follows.

In the first phase, the MSRE started with fresh fuel, accompanied by fission product removal. In the second phase, U-233, bred from Th-232, was introduced as a new fuel source, demonstrating the reactor's transition to a thorium-based cycle. Finally, the reactor operated with a mixed fuel composition of U-235 and U-233, with periodic noble gas removal to maintain a sustainable system. This multi-phase approach provided insights into fuel reprocessing, reactivity management, and the potential benefits of thorium-based reactor systems.

This design also focused on a detailed isotopic analysis of special nuclear materials such as Uranium and Plutonium isotopes to further investigate their quality and quantity. Additionally, by analyzing the relationship between the neutron spectra generated by the MSRE and the cross-sections of specific materials, the understanding of MSRE performance phenomena was enhanced.

In conclusion, this research enhanced the understanding of the features and characteristics of MSRE, leading to the potential application of new safeguards methodologies. Furthermore, this research was planned to investigate safeguards protocols by modeling and integrating appropriate detectors into the reactor system.

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Keywords: MSRE, Simulation, Serpent, Safeguards-By-Design (SBD), Nuclide composition

Comparison Study on MBE for Uncertainty Expression Methods in Benchmark Fuel Fabrication Plant

Hyun Cheol Lee, Jung Youn Choi, Hyunju Kim, Jinho Ryu, and Haneol Lee*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*haneol@kinac.re.kr

The International Atomic Energy Agency (IAEA) uses Material Balance Evaluation (MBE) to assess the nuclear material accounting of nuclear facilities. The Material Unaccounted For (MUF) is defined as the difference between the book inventory and the physical inventory. The IAEA's MBE evaluates the significance of the MUF by comparing it to its uncertainty, using statistical inference. The IAEA applies a top-down approach to MBE, which minimizes the use of detailed facility-specific accounting information. However, this approach may introduce statistical bias as it does not account for the unique characteristics of each facility's accounting system.

In this study, two alternative uncertainty evaluation methods for the MBE were investigated. The first method was the Guide to the Expression of Uncertainty in Measurement (GUM) established by the International Organization for Standardization (ISO). The GUM method propagated all components of uncertainty of individual measurement, whereas the IAEA method homogeneously treats items within the same stratum.

The second method was the Monte Carlo (MC) method, which generated the relative standard deviation (RSD) for all measurement results in a facility. The MC method calculated the MUF for the inventory with RSD and we repeated the process 500 times. After that, the average and uncertainty calculated for the repeated calculation result and the MUF and MUF uncertainty were defined.

For the comparison of the three methods, a hypothetical bulk handling facility where benchmarked a fuel fabrication plant in the ROK was considered. The accounting system of the benchmark facility consists of weighing, sampling, element concentration and isotopic ratio analysis. It was assumed that the uncertainty of the accounting system was same as the ITV 2010.

The results showed that the IAEA method evaluated σ_{MUF} as 37.58 kg, the GUM method yielded u_{MUF} as 36.65 kg, and the Monte Carlo method produced σ_{MUF} of 45.97 ± 0.79 kg at a 95% confidence level. While the IAEA method is applicable to all facilities, it struggles with precise measurement uncertainty evaluation. The GUM method, although requiring detailed intrinsic uncertainty data, allows for a more comprehensive evaluation of measurement uncertainties. The MC method is useful even with limited intrinsic uncertainty information, though it is less quantitative than the GUM method.

Given the inevitability of MUF, accurate uncertainty evaluation is crucial. For a quantitative MUF evaluation in national safeguards inspections, it is necessary to supplement the IAEA method by appropriately applying the strengths of both the GUM method and the MC method.

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Keywords: MBE, MUF, Measurement uncertainty, IAEA, GUM, Pseudo random number

Comparison of Safeguards Implementation for Radioactive Waste Containing Nuclear Material

Yunhui Kim and Ickhyun Shin*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*ihshin@kinac.re.kr

Under the Republic of Korea-IAEA Comprehensive Safeguards Agreement (CSA), all nuclear materials (NM) used in nuclear activities are subject to safeguards, requiring the reporting of the quantity and location of NM. However, applying safeguards to all radioactive waste containing NM is inefficient from an inspection standpoint. Therefore, the IAEA allows for the mitigation of safeguards burdens based on the content and characteristics of each radioactive waste by means such as termination of safeguards on measured discard, exemption from safeguards, and transferring waste into retained waste. The IAEA allows the termination of safeguards on “Nuclear material upon a determination by the IAEA that the material has been consumed, or has been diluted in such a way that it is no longer usable for any nuclear activity relevant from the point of view of safeguards, or has become practically irrecoverable”. Accordingly, wastes with concentrations lower than the maximum concentration for termination of safeguards are subject to the safeguard termination. The maximum concentration standard is determined by comparing the relative effort required to recover NM from the waste with the effort to extract NM from ore. Next, exemption from safeguards can be granted when “NM is recoverable and used for non-nuclear activities under Article 13 of INFCIRC/153, or when the quantity of NM is less than the IAEA exemption criteria specified in Article 37 of INFCIRC/153”. Although exempted nuclear material is not subject to safeguards, it is still subject to reporting under the Additional Protocol (AP). Finally, radioactive waste containing NM can be transferred into retained waste for temporary storage. Retained waste is “NM generated from processing or from an operational accident, which is deemed to be unrecoverable for the time being but which is stored”. When radioactive waste containing NM is transferred into retained waste, it remains subject to safeguards but is not subject of NM accountancy. The choice of safeguard implementation method depends on the characteristics of each waste, such as the kind of NM it contains, the amount of NM, and its storage location. This paper aims to compare the safeguard implementation method to provide operators with guidelines for the proper management of radioactive waste containing NM. In particular, comparing several conditioning methods such as incorporating into a ceramic matrix, incorporating into a vitrified matrix, Macro-encapsulating, Micro-encapsulating, etc for diluting NM in waste to terminate safeguard is dealt with.

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Keywords: Termination, Exemption, Retained waste, Nuclear material, Radioactive waste

Analysis of Safeguards Measures for Detection of Misuse of Decommissioning Nuclear Facilities

Ickhyun Shin, Myungtak Jung, Yunhui Kim, and Seungmin Lee*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*seungmin@kinac.re.kr

To detect any diversion of nuclear material and any misuse of nuclear facilities, the IAEA applies Safeguards measures based on the nuclear material accountancy such as physical inventory verification. The containment and surveillance (C/S) measures are also complemented to maintain the continuity of knowledge of Safeguarded nuclear materials and facilities.

The Safeguards measures including the C/S are not only applied to the operating nuclear facilities but also the decommissioning facilities. But the Safeguards measures based on the nuclear accountancy are no longer applied to the decommissioning facilities after all the nuclear materials are removed in the facilities. And only the design information verification (DIV) activities and C/S measures will be remained to detect the misuse of essential equipments (EE) such as reactor core, steam generator, pressurizer. Since the decommissioning facilities can restart the operation of the reactor by introducing undeclared nuclear materials, the C/S measures play important role to ensure that the EEs have never been restarted after the last DIV.

However, it is difficult to apply C/S measures to decommissioning facilities in the same way as operating nuclear facilities. In particular, in the case of nuclear power plants, it is difficult to install surveillance cameras or seals like operating nuclear power plants because large objects such as RCP and steam generator are frequently moved through the existing strategic points (e.g., canal gate and equipment hatch) during the decommissioning process. In addition, existing strategic points must be newly devised because there will be new paths that can transport nuclear materials or essential equipments.

In this paper, C/S measures applicable to decommissioning nuclear power plants from which nuclear materials have been removed or alternative measures if they are not applicable were sought by analyzing scenarios for misuse of nuclear facilities suggested by the IAEA SG-GD-12278 (2014-06-18). As a result, attaching seals directly to the reactor vessel is the most effective. But it is sometime difficult to apply seals on the reactor vessel because of high radioactivity. Alternatively, if the missile shield is installed on the top of the reactor vessel, the reactor head can be sealed by sealing the missile shield, so the same effect as sealing the reactor itself can be achieved. Several records can also be alternatives for C/S measures. Reactor operation record and steam generator water flow rate record, if record keeping systems are still applicable, can be provided to assure that the reactor was not operated.

In the future, a feasibility study is needed to determine whether the Safeguards measures derived from the results of this study are applicable in the actual decommissioning facilities.

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Keywords: Containment and surveillance, Safeguards measures, Decommissioning facilities

Considerations on Application of Emerging Technologies to Safeguards

Kyunglee Kang, Wooseub Kim, and Hana Seo*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hana@kinac.re.kr

The IAEA, which was established to prevent the proliferation of nuclear weapons, should conduct effective and efficient safeguards within its limited resources such as budget, human capacity, etc. In light of these constraints, the IAEA has sought to optimize resources and enhance the efficiency of its safeguard activities by incorporating emerging technologies, such as Artificial Intelligence (AI), Machine Learning (ML), and robotics. This article evaluates the IAEA's approach to determining the appropriate timing for the application of new technologies, emphasizing the importance of maintaining the integrity of IAEA safeguards.

As outlined in the IAEA's Mid-Term Strategy, Resource Mobilization Priorities, and Development and Implementation Support Programme, the IAEA has explored AI/ML, robotics, and remote/unattended technologies. Although the concepts behind these technologies were proposed over 50 years ago, they have only been adopted for application within the past decade. While these technologies offer significant potential for enhancing verification efficiency, it is essential to establish an environment that supports their use. Also, there should be measures to prevent the situation that it is used for other purposes. In some cases, it is challenging to guarantee that the operation of facilities will not be affected. The application of these technologies has thus been determined based on specific conditions.

This article identified the conditions that influenced the IAEA's decision-making process regarding the adoption of particular technologies. Furthermore, it assessed the necessary feasibility studies that should be conducted prior to the promotion of research and development, in contrast to the approach of immediately applying new technologies that can improve safeguard efficiency.

Keywords: Nuclear, Safeguard, R&D, Feasibility study, Adoption moment of new technology

Analysis of the Requirements for Transitioning of Nuclear Material Accountancy Report

Seungmin Lee and Hyun-jin Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hjkim@kinac.re.kr

Under the ‘ROK-IAEA Comprehensive Safeguards Agreement’, the Republic of Korea (ROK) is required to submit the accounting reports on all nuclear material movements and inventories to the International Atomic Energy Agency (IAEA). Currently, these reports are prepared using a fixed format based on the Subsidiary Agreement with the IAEA. However, this format, which was developed in the 1970s, has significant limitations for the preparation of nuclear material accountancy reports. For example, the nuclear material batch name must be exactly eight characters long and must be positioned between columns 13 and 20 of the report. In addition, the existing format does not support the inclusion of additional information, such as the origin of the nuclear material, and is incompatible with the latest information technology standards. These limitations not only hinder the accurate and efficient reporting of nuclear materials but also pose challenges in integrating modern data management systems.

To address these limitations, the IAEA has proposed a new reporting format: the XML format, which has many advantages. The XML format allows an unlimited number of characters for the names and quantities of nuclear material, making it possible to manage a larger amount of information simultaneously. However, given that all of the ROK’s nuclear material accountancy reports are currently in the fixed format, careful consideration is required before adopting the new XML format. This paper analyses the new XML format and reviews international case studies to identify the necessary requirements. The analysis concludes that a redesign of the database structure is required to change the format and manage the increased information required. Furthermore, the transition to the XML format will necessitate comprehensive training for personnel to ensure a smooth adaptation to the new system. In addition, as current regulations and guidelines are based on the current fixed format, new documents need to be developed. The results of this study will be applied to the design of a new database structure for the management of all nuclear materials using the XML format proposed by the IAEA.

Acknowledgements

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Keywords: Safeguards, Nuclear material accounting reports, XML format, Database structure

Proposed Improvements to the Small Quantity Nuclear Material Accounting and Control (NMAC) Online System

Chul Heo* and Myungtak Jung

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hc0228@kinac.re.kr

The ROK submits nuclear material accounting reports to the IAEA by Article 63 of the CSA and Article 98 of the Nuclear Safety Act. Based on the aforementioned legal basis, operators in the ROK prepare these reports and submit them to KINAC through the State Declaration Portal. Regarding Location Outside Facilities (LOFs), KINAC prepares and submits the accounting reports. The efficient operation of an online system is essential to compile and organize reports from the 256 institutions holding small quantities of nuclear material as of the end of 2023. This paper proposes three directions for improving the small quantity NMAC online system currently operated by KINAC. Firstly, for the online system to accurately generate physical inventory lists and material balance reports, it must precisely verify exemption status as of a specific date. However, current errors necessitate manual preparation of these reports, highlighting the need for the online system improvements. Secondly, some data registered in the online system include logical inconsistencies. A comparative analysis with the Korea Safeguards Information System (KSIS) is needed to address these issues, as the data reported to the IAEA via KSIS contain no such errors. Using KSIS data as a reference is expected to resolve these logical errors. Lastly, entering a single inventory change in the online system requires seven steps of information, posing significant challenges for users unfamiliar with the online system. The online system should be modified to minimize the required input fields and simplify data entry to enhance user convenience. Additionally, transitioning from a text-based system to a graphical user interface (GUI) will make it easier for new users to input data. In conclusion, for the online system to efficiently manage reports from LOFs, it must be user-friendly and convenient. Improvements in exemption history management, correcting logical data errors, and introducing a GUI will likely increase user satisfaction. Securing the necessary budget and personnel to implement these improvements, along with continuously refining the online system based on user feedback, will significantly aid in fulfilling the accounting responsibilities of LOFs.

Keywords: Safeguards, Accounting reports, LOFs, Online system

An Analysis of the Revision History for the Policy on Nuclear Material Accountancy at KAERI

Jinha Choi*, Hyun Jo Kim, and Ju-Ang Jung

Korea Atomic Energy Research Institute (KAERI)

*Choi.jh@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI), established in 1959 as the first national research institution in South Korea, performs comprehensive research and developments in atomic energy to contribute to academic advancement, energy security, and the promotion of nuclear energy utilization.

The institute consists of 11 facilities that use nuclear materials and carry out various research activities. To comply with the obligations under domestic law and efficiently manage nuclear material accountancy within the institute, it operates the ‘Regulation on Accounting of Nuclear Materials’ as a guideline.

This paper analyzes the revision history of KAERI’s nuclear material accounting policy since its establishment.

The policy is prepared and is subject to approval by the Nuclear Safety and Security Commission (NSSC) by the notice, ‘Regulation on the Preparation of Policy on Accounting of Specific Nuclear Materials’. In the past, detailed guidelines for the notice required specific information such as the authorized quantity of nuclear materials and facility layout diagrams. As a result, whenever facility information was changed, a revision to the policy was required. KAERI has received approvals for the revised policy almost annually since 2004.

In 2014, the detailed guidelines for the regulations were revised to exclude detailed facility information in policy, the policy took a form similar to the present. Additionally, the national inspection system is changed to focus on verifying whether nuclear material accountancy is properly implemented upon the approved policy, independently of the IAEA. Accordingly, KAERI has revised its policy to reflect follow-up actions based on national inspections.

This year, the KAERI’s policy on nuclear material accountancy is revised. The main changes include (1) the application of recommendations from national inspection, (2) the addition of a new facility code, and (3) a new substitute condition for internal education. In addition, the revisions clarified the names of related institutions and guidelines to prevent confusion.

In the future, if there are any necessary revisions to related notices for safeguards or nuclear material accountancy, KAERI will actively provide opinions and ensure efficient internal nuclear material accounting through proactive revision of the policy.

Keywords: Revision of the policy nuclear material accounting, Safeguards, Domestic nuclear safety law, Public notice

Study on the Establishment and Operation of the International Expert Committee for the International Peer Review of i-SMR Safeguardability

Jae Min Lee, Soo Hwan Lee, Saerom Son, and Eun-ha Kwon*

Mirae Consulting Group

*kwoneh@mcg.re.kr

About 80 Small Modular Reactor (SMR) models are under development worldwide, and South Korea is also developing an i-SMR. It is being designed to ensure safety and peaceful use from a nuclear non-proliferation perspective, taking into account the Safeguards-By-Design (SBD) concept. i-SMR Safeguards System developed in the first phase will undergo a peer review by an international expert committee to ensure external credibility. To facilitate this, the forms of existing international expert committee related to SMRs or Safeguards have been investigated and analyzed.

As a result, it was confirmed that it can be carried out in three main forms. The first approach is to establish an international expert committee using international organizations. For example, technical working groups of international organizations related to SMRs include the IAEA TWG-SMR, IAEA SMR Regulators' Forum, and the OECD/NEA Expert Group on Small Modular Reactors (EGSMR). The second approach is to independently establish an International Expert Review Committee and publish the results in the form of a Review Report. This method is typically used to evaluate the overall research output of an organization or to set its strategic direction. The third approach is to use ongoing panels in domestic public organizations to review the results of this study. Examples that can be used include the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the Generation IV International Forum (GIF) Proliferation Resistance & Physical Protection Working Group (PRPPWG), Member States Support Programmes (MSSP) with cooperation from KAERI.

In order to organize the most ideal international expert committee to meet the objectives of this study, the following criteria were set for the selection of the method of organization: 1) whether a report can be published by a reputable organization (Reliability), 2) whether there is a Counterpart who can cooperate (Accessibility), and 3) whether the cooperation can be carried out quickly (Immediacy). After applying these criteria to form the most ideal international review committee, the final report is expected to be published.

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Keywords: i-SMR, Safeguardability, International peer review

International Efforts to Address the Interface Between Safety, Security, and Safeguards in Small Modular Reactor Design: Progress at the IAEA

Gaeul Choi and Sungyeol Choi*

Seoul National University (SNU)

*choisys7@snu.ac.kr

There has been growing global interest in small and medium-sized or modular reactors, as well as reactors deploying new design concepts, due to their ability to meet the need for flexible power generation, achieve net-zero carbon goals, and enhance safety performance through inherent and passive safety features. Small Modular Reactors (SMRs), which produce up to 300 MW(e) per unit (while conventional nuclear reactors typically generate three times more), have emerged as a promising solution to these challenges. Additionally, various reactor concepts currently under research and development do not use light water as a coolant — so-called non-Water Cooled Reactors (non-WCRs) — such as molten salt reactors or high-temperature gas-cooled reactors. Some countries are even considering marine-based floating nuclear power plant concepts.

Currently, more than 80 commercial SMR designs are being developed worldwide, targeting a range of outputs and applications, with some designs claimed to be near-term deployable. Due to the uniqueness of the new reactor design in terms of technology, safety characteristics and capacity etc., it has been emphasized that new international standards are needed to address the new challenges and considerations stemming from this uniqueness. The interface between Safety, Security, and Safeguards (3S) is one of the key issues as it can create synergies between the three disciplines from the very initial design stages. This includes optimal design solutions, fuel design and manufacturing elements, regulatory frameworks focused on integrating safety, security, and safeguards, nuclear material accountancy, and access control.

As many SMRs and non-WCRs are still in the early stages of development, there is an opportunity to comprehensively integrate safety, security and safeguards into their design—a concept known as 3S by design (3SBD). The International Atomic Energy Agency (IAEA), as the global focal point for nuclear-related activities, has stated that, in the long term, it might explore integrating certain activities related to safeguards, safety, and security, potentially creating synergies and efficiencies. The agency has therefore been actively providing a platform to discuss challenges and share experiences related to the development and future deployment of SMRs and continues its efforts toward establishing an international standard.

This paper addresses recent progress in the discussion of the 3S concept in SMR and non-WCR design, highlighting the ongoing initiatives and the current efforts being undertaken at the IAEA. Additionally, it suggests key takeaways that can be useful for the advancement and integration of SMRs in Korea.

Keywords: Small modular reactor, Non-water cooled reactors, Interface between safety, Security and safeguards, 3S by design

The Values of Strengthening the International Nonproliferation Regime With Additional Protocol From a Capacity Building Perspective

Jae-Jun Han* and EunBee Park

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jjhan@kinac.re.kr

This paper analyzes the role of the International Atomic Energy Agency (IAEA) member states in capacity-building with the Additional Protocol (AP) for strengthening the international nonproliferation regime. The AP is a tool designed for the IAEA verification, enabling more effective monitoring and evaluation of states' nuclear-related activities in addition to the Comprehensive Safeguards Agreement (CSA) under the Nuclear Nonproliferation Treaty (NPT) regime. With the backdrop of energy security and carbon neutrality, nuclear power has become increasingly important around the globe; accordingly, the importance of promoting IAEA Safeguards capacity-building programs for the embarking and developing states is worth investigating, particularly in terms of effectively implementing the CSA and AP. Against arguing the concerns about sovereignty infringement, the advantages of leading the momentum, such as education and training, are discussed from a state standpoint. The values are investigated from the perspectives of global leadership, national and regional security, diplomatic trust-building, and expanding the economic potential of nuclear energy.

Firstly, by leading capacity-building efforts, the state can demonstrate its leadership within the global nonproliferation regime, thereby enhancing its position and role in nuclear governance. Further, it solidifies international trust in the state, boosting opportunities for interstate cooperation. Simultaneously, facilitating IAEA-driven initiatives fosters the exchange of peaceful nuclear technologies among states. This is particularly beneficial for providing technical support or educational opportunities to developing countries, thereby increasing leading states' international influence. Secondly, as the AP strengthens the global nonproliferation regime, the state can contribute to reducing nuclear risks in the region. Adopting the Additional Protocol by neighboring countries would provide the state with greater transparency and information on regional nuclear activities, contributing to national security and regional stability. Lastly, promoting capacity-building of the Additional Protocol can yield long-term economic benefits for the state. By actively supporting international nonproliferation norms, the state can strengthen its image as a responsible and peaceful supplier, thereby fostering international trust in its nuclear-related industries and ultimately facilitating access to global markets.

Keywords: Nuclear nonproliferation, Capacity building, Comprehensive safeguards agreement, Additional protocol

Identification of the International Nonproliferation and Security Academy's Strength to Share Its Experience of Operating a Nuclear Security Training and Support Centre

EunBee Park* and Jae-Jun Han

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*ebpark@kinac.re.kr

The International Atomic Energy Agency (IAEA)'s International Network for Nuclear Security Training and Support Centres (NSSC Network) was created in 2012 to facilitate collaboration among nuclear security and training and support centres. The NSSC Network Database allows member institutions to voluntarily share their status and experience, forming a collective dataset of useful references for other institutions undergoing a similar stage or situation while operating their nuclear security training and support center. The Republic of Korea's nuclear security training and support centre, the International Nonproliferation and Security Academy (INSA), actively participated in the NSSC Network cooperation from its planning stage of establishment.

As global nuclear security faces unprecedented threats, the importance of building nuclear capacity through broader engagement and collective effort has been highlighted, especially at the IAEA International Conference on Nuclear Security (ICONS) 2024. Since its establishment in 2014, INSA has contributed to this course of efforts for reinforcing nuclear security capacity by developing and providing education and training programs targeting both domestic and international trainees. As 2024 marks INSA's 10th anniversary, it entered the maturation stage where it can take an expanded role within the international community rather than focusing only on its program. One of the possible roles to be developed would be to be an advisor for its partners interested in nuclear security capacity building and even to establish their own nuclear security training and support centre.

As the preliminary study, this paper analyzed the INSA's status compared to INSA's major partner countries that showed interest in participating in INSA's International Training Courses (ITC). By doing so, the comparative data illustrates INSA's position among its partners and identifies its strength as a leading NSSC. The data revealed that the areas of expertise that INSA can provide are in the areas of feasibility and planning and human resources development. Specifically, INSA can assist its partner institutions in project management, strategy development, and human resources development training activities.

Keywords: Nuclear nonproliferation and security, Capacity building, Education and training, International cooperation

A Research on Validation and Optimization of Obligation Tracking System

Hee Su Choe* and Hansol Ko

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hschoe@kinac.re.kr

Korea currently has nuclear cooperation agreements (NCA) with 28 countries, some of which have administrative agreements (supplementary agreements), which require annual reporting of domestic inventories and management of imports and exports after fulfilling bilateral procedures. To ensure effectiveness and minimization of human error in implementing the NCA, a database of obligated items was established in 2021, and in 2023, an Obligation Tracking System (OTS) module was also developed to provide integrated management of item monitoring and related regulatory information and to support the preparation of annual reports. In order to verify and validate the practical applicability of the developed OTS in fulfilling export control obligations under the NCA and to examine improvement needs, this paper presents the results of the development of representative scenarios and the integrated validation of the OTS.

The scenarios were created considering item classification, obligated country type, and domestic or international transfer types. In particular, to optimize the OTS's practical applicability, cases implemented within the last three years and with plans to re-transfer in the future were selected and used as representative validation targets. As a result of the scenario test, the main requirements for optimization were identified as information linkage with NEPS, inputting the information difference between shipment and receipts quantity, ability to utilize existing inventory change reporting file, and input data validation function.

The results were reflected in improving and optimizing OTS functions, divided into error correction, improvement, and development of new functions. In particular, as a new function expected to be utilized by domestic users, the function to link regulatory export control information with NEPS was added. In addition, the inventory change input function using the ICR data was implemented to allow users to upload with some modifications in the file, considering the inconsistency with the origin management standards in the current domestic system and the current situation that separates the system's operation.

In conclusion, the optimization requirements for the most important 'item inventory management and information linkage' feature of the OTS derived from the system integration validation were completed before the pilot test. In the future, it is expected that the OTS will be more user-friendly and easier to use by adding some limited functions, reflecting the opinions of the pilot test by the actual users of the item-holding organizations.

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Keywords: Export control, Nuclear cooperation agreement, Obligation tracking system

A Comparative Analysis of the Compliance Programs in South Korea and the U.S.

Hansol Ko* and Hee Su Choe

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hsko@kinac.re.kr

This study provides a comparative analysis of South Korea's Compliance Program (CP) for strategic items and the United States' Export Compliance Program (ECP), focusing on their similarities and differences. A Compliance Program refers to a system in which companies, universities, and research institutions establish independent export transaction review bodies to assess strategic item exports according to operational regulations. This system involves either rejecting export transactions or applying for professional judgment and export licenses from the relevant authorities. The study aims to examine the background and operational methods of the CP systems in South Korea and the U.S. from various perspectives.

Both programs share a common goal of safeguarding national security by preventing unauthorized exports of strategic items. The CP system plays a crucial role in ensuring compliance with export control procedures and regulations by providing companies with benefits such as expedited export control processes, reduced legal risks, and enhanced international credibility.

However, there are differences in the legal foundations and operational approaches of the two systems. South Korea's CP was introduced in 2007 under the Foreign Trade Act and is managed by the Ministry of Trade, Industry, and Energy (MOTIE) and the Korea Strategic Trade Institute (KOSTI). This system was implemented to promote voluntary compliance with legal regulations among companies, aligning with South Korea's trade promotion policies. In contrast, the U.S. ECP was established in 1979 under the Export Administration Regulations (EAR) and the International Traffic in Arms Regulations (ITAR) and is managed by the Bureau of Industry and Security (BIS) and the Department of State (DOS). The ECP serves as a tool for enforcing strict legal regulations in the U.S.

In conclusion, the CP systems in South Korea and the U.S. have been developed and operated differently, reflecting each country's environment and national priorities. South Korea's CP focuses on enhancing export autonomy by encouraging companies to establish internal organizations and management systems for compliance. In contrast, the U.S. ECP emphasizes strengthening international competitiveness through strict legal protections and regulatory compliance, prioritizing national security over export promotion. Companies that comply with the ECP gain high trust from both the U.S. government and the international community in terms of regulatory compliance and national security. By establishing CP systems that align with their respective environments and priorities, both South Korea and the U.S. provide advantages to their exporting companies while ensuring compliance with strategic item export controls.

Keywords: Strategic items, Export control, Compliance program, Legal regulation

Analysis of Recent Operational and Maintenance Trends at Nuclear Facilities in the DPRK Yongbyon Nuclear Complex

Gayeon Ha, Dongjin Kim, Byungmarn Koh, and Hana Seo*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hana@kinac.re.kr

This study, based on satellite imagery and publicly available information, analyzed the operational trends of the nuclear reactors (5 MWe reactor, Experimental Light Water Reactor (ELWR)) and the Radiochemical Laboratory (RCL) at North Korea's Yongbyon nuclear complex. It also considered the implications of the maintenance of the RCL's coal-fired thermal plant observed from March to June 2024. The study began by outlining the role and layout of the RCL and its thermal plant within North Korea's nuclear program. To estimate the historical operational cycles of the reactors and the RCL, it examined signs of facility operations observed in satellite imagery, such as steam emissions, vehicle movements, and structural changes. Our analysis suggests that the recent maintenance at the thermal plant may have been preparatory work to support the reprocessing of spent nuclear fuel from the Yongbyon reactors. Ongoing monitoring of these facilities could provide valuable insights into North Korea's nuclear activities and technical capabilities.

Keywords: Yongbyon nuclear complex, Radiochemical Laboratory (RCL), Thermal plant, Satellite imagery

Research on Unstructured Data Classification Schemes for Monitoring Nuclear Activities

Yonhong Jeong* and Gayeon Ha

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jyh1404@kinac.re.kr

The international community recognizes North Korea's nuclear activities as one of the significant security threats. North Korea has demonstrated its nuclear capabilities through numerous nuclear tests and missile launches, posing a severe threat to the peace and stability of not only its neighbors but also the entire world. To effectively monitor and respond to this threat, the international community has used a variety of methods to monitor and analyze North Korea's nuclear activities.

Traditional methods of monitoring nuclear activities include analyzing satellite imagery, detecting seismic waves, and analyzing nuclides by capturing inert gases. Each method has limitations and advantages, and the need for a data platform that integrates and analyzes these methods is emerging. In particular, recent advances in artificial intelligence (AI) and big data technologies are opening up new possibilities for efficiently processing and analyzing massive amounts of data.

Integration of various data sources: Integrating various data sources, such as satellite imagery, seismic data, and nuclide information, enables comprehensive analysis and provides more accurate information about North Korea's nuclear activities.

Real-time data processing: Artificial intelligence and big data technologies can be used to build a system that can process and analyze massive amounts of data in real-time. It will enable rapid detection and response to North Korea's nuclear activities.

Automated analysis and warning system: Developing a system that can automatically analyze anomalies related to nuclear activity and issue warnings. It will improve the efficiency of nuclear activity detection and enable rapid response.

Enhance international cooperation: The data platform will enhance cooperation with the international community. It enables various countries and organizations to share data and information and work together to respond effectively to the North Korean nuclear threat.

Data distribution can enhance and augment big data platforms. However, for such data distribution, it is necessary to precisely define the additional information that data providers should provide to consumers and re-categorize the metadata of unstructured data collected from each data source into additional information. To this end, this study aims to define a taxonomy for unstructured data and derive metadata to standardize additional attributes.

Acknowledgements

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Keywords: Nuclear non-proliferation, Data platform, Methodology of monitoring nuclear activities

Investigation and Analysis of Unstructured Data Collection and Processing Techniques for Assessing Nuclear Proliferation Risk

Jae Min Lee, Soo Hwan Lee, Saerom Son, and Eun-ha Kwon*

Mirae Consulting Group

*kwoneh@mcg.re.kr

Northeast Asia is a region where the interests of the world's major powers are inextricably linked, and various conflicts among regional countries arise from historical and territorial issues. Recently, neighboring countries have been engaged in nuclear energy activities, which necessitates monitoring to prevent these activities from leading to nuclear proliferation for military purposes. To address this, developed a nuclear proliferation risk assessment model in this study. However, since physical access to internal information for detecting nuclear activities in neighboring countries is extremely limited, it may be more realistic to use publicly available open source. Such Open-source intelligence (OSINT) exists in large and varied forms, so systematic data collection, processing, and analysis techniques are essential for effective analysis.

Nuclear proliferation risk assessment model generally evaluates risks based on technical factors and political factors. As reliable data must be collected to assess proliferation risk, this study investigated recently developed data collection, processing, and analysis (especially classification) techniques to apply optimal methodologies. Data collection methods typically include web crawling, which automatically collects data from various websites and Open APIs, which allow real-time data retrieval from data providers. To analyze the collected data, an effective data processing workflow is required. This usually involves the use of Natural Language Processing (NLP). Recently, several dynamic word embedding techniques have been developed, such as Bidirectional Encoder Representations from Transformers (BERT), Large Language Model Meta AI (LLaMA), and Robustly optimized BERT approach (RoBERTa). These models are based on the Transformers architecture using the Self-Attention mechanism, which replaces previously used algorithms such as Word2Vec, Global Vectors for Word Representation (GloVe), and Recurrent Neural Networks (RNNs) and enables training on large datasets. In the future, these technologies will be used for data processing to perform classification for nuclear proliferation risk assessment.

This involves the investigation of methodologies for real-time collection of unstructured data from various sources, as well as processing techniques for refining and preprocessing the data to remove irrelevant information and extract key insights, thereby converting it into an analyzable format. Subsequently, machine learning-based classification techniques are applied to categorize data to collect reliable data for nuclear proliferation risk assessment.

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Keywords: Unstructured data, Nuclear proliferation risk, Open source, NLP, LLM

Research on the Development of the Nuclear Proliferation Risk Assessment Model

Taeyeong An and Seung Min Woo*

Kyung Hee University

*woosm@khu.ac.kr

The Northeast Asian region, where the Korean Peninsula is located, is entangled in the complex interests of global powers such as the United States, Russia, and China. Ongoing nuclear activities in the region are contributing to heightened security concerns. Therefore, it is necessary to continuously monitor these nuclear activities to detect signs of nuclear proliferation in neighboring countries in advance and respond proactively, ensuring that peaceful nuclear activities do not evolve into military-related nuclear proliferation. However, detecting nuclear activities in neighboring countries from outside is extremely challenging due to the limited physical access to internal information. As a result, there is need to develop a deep learning-based nuclear proliferation risk assessment model in neighboring countries by utilizing publicly available information, which is relatively easier to obtain.

The nuclear proliferation risk assessment model aims to collect, process, and analyze publicly available information related to neighboring countries' nuclear activities of neighboring countries, and integrate this data with a deep learning model to analyze various aspects of nuclear proliferation stages and intuitively represent the risk of nuclear proliferation.

In the nuclear proliferation risk assessment model, the evaluation factors are divided into technical and political factors. The technical factors are primarily classified into nuclear materials, nuclear facilities, and nuclear technology. The political factors are influenced by each country's non-proliferation system, international policies, and domestic policies. For Assess the risk of nuclear proliferation, a formula was developed using technical and political factors.

Moving forward, the concept of Nuclear Latency will be applied to link the stages of nuclear proliferation with the calculated phases. This will allow for a quantitative analysis and evaluation of each stage of nuclear proliferation. And through expert consultations, the model will be further developed and refined by incorporating weighted factors into the existing formula. By assigning appropriate weightings to the various variables and factors, the goal is to enhance the model's precision and reliability.

Acknowledgements

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Keywords: Nuclear proliferation, Risk assessment model, Nuclear latency, Deep learning

Comparative Analysis of Technological Developments in Nuclear Non-Proliferation Assessment Methodologies

Ju-Ang Jung*, Jinha Choi, Hojin Hwang, and Hyun-Jo Kim

Korea Atomic Energy Research Institute (KAERI)

*jajung@kaeri.re.kr

The development of nuclear energy systems is experiencing significant advancements globally, with particular attention to ensuring the non-proliferation of nuclear materials. As the international community intensifies efforts to enhance safeguards, the necessity of robust evaluation methodologies becomes evident, especially for next-generation nuclear reactors, and Small Modular Reactors (SMRs). This study aims to analyze and compare the development of nuclear non-proliferation assessment methodologies between domestic and international frameworks, focusing on the technological advancements and regulatory standards.

Globally, key players such as the Generation IV International Forum (GIF) and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) have set ambitious goals for non-proliferation in new nuclear technologies. The evaluation of proliferation resistance has become a critical aspect of nuclear safety and security. In particular, the deployment of SMRs introduces new challenges and opportunities in non-proliferation assessment, as their modularity and operational flexibility require tailored safeguards approaches. International efforts have resulted in the development of general principles and technical standards to address proliferation resistance in both present and future reactor designs, with a focus on ensuring that SMRs meet these stringent requirements.

This research explores the suitability of different non-proliferation assessment methodologies for SMRs, offering insights into how these methodologies can be adapted to effectively monitor and secure the unique features of SMRs. The comparative analysis underscores the alignment and divergence between domestic and international strategies, providing a clear perspective on how the international community can work together to enhance the transparency and effectiveness of nuclear safeguards for SMRs.

Keywords: Proliferation resistance, IAEA safeguards, Non-proliferation assessment methodology, GIF, INPRO, SMR

Fire Protection Program for Nuclear Fuel Cycle Facilities and Radioactive Waste Management Facilities

Naon Chang*, Yoon-Do Oh, HeungJu Cho, Dong-su Kim, and Tack-Jin Kim

Korea Atomic Energy Research Institute (KAERI)

*cno0622@kaeri.re.kr

Nuclear facilities must implement various safety measures during operations in compliance with the Nuclear Safety Act. These safety measures include the measures regarding radiation exposure, facility inspections, and safety operation. They are related to fires in the facilities in terms of design and operation. Therefore, a fire protection program must be established and implemented in accordance with the relevant regulations to ensure the safe operation of nuclear facilities. In this study, the regulatory requirements for fire protection in nuclear fuel cycle facilities and radioactive waste management facilities were compared. In addition, the cases of implemented fire protection programs were analyzed.

The fire protection program is a plan for the equipment, procedures, and human resources required to perform fire protection activities. The fire protection program is composed of a fire hazard analysis and a fire protection action plan. The fire risk analyses for nuclear fuel cycle facilities and radioactive waste management facilities are necessary to be conducted in accordance with the Nuclear Safety and Security Commission Notices. In particular, for radioactive waste management facilities, the fire risk analysis must be performed every 10 years from the operation commencement date. The fire protection action plan is composed of the status of facility, fire protection organization, fire prevention and response, test procedures for fire protection system, including firefighting and evacuation equipment. The fire protection action plan for nuclear fuel cycle facilities is established depending on the Nuclear Safety and Security Commission Notices No. 2020-2. In accordance with the Notice No. 2020-2, nuclear fuel cycle facilities must undergo an implementation status inspection by external experts once every three years. On the other hand, fire protection action plan for radioactive waste management facilities is governed by Rule of the Nuclear Safety and Security Commission No. 50. This rule does not include the provisions for inspection by external experts. At the nuclear fuel cycle facility and radioactive waste management facility in KAERI, the fire protection action plans were established depending on the regulatory regulation. The training for the fire protection organization, test and maintenance of equipment are conducted periodically at KAERI based on the fire protection action plan.

In this study, the fire protection programs for nuclear fuel cycle facilities and radioactive waste management facilities were analyzed. It was confirmed that the fire protection programs can be different depending on the type of license. These results can be useful for managing the fire protection programs for nuclear facilities.

Acknowledgements

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Keywords: Fire protection program, Nuclear fuel cycle facility, Radioactive waste management facility, Requirement

Assessment of Security Performance Indicators for the Implementation of Risk-informed and Performance-based Security Regulations in Nuclear Facilities

Hyunjoo Lee* and Taehee Kim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*sn220865@kinac.re.kr

Nuclear facilities must safely manage spent nuclear fuel generated during their operation and decommissioning processes. As security regulations have become increasingly significant, efforts to introduce regulations integrating safety and security have continued. The U.S. Nuclear Regulatory Commission (NRC) has implemented a Reactor Oversight Process (ROP), a risk-informed and performance-based regulatory system that oversees both safety and security by utilizing limited resources efficiently.

In the Republic of Korea (ROK), research has been conducted to apply this effective oversight system. However, due to the dual regulatory system separating safety and security, the development of the ROP for domestic nuclear regulation has progressed differently in these areas. In the safety regulation sector, ROK has developed a domestic safety ROP framework, benchmarked against the NRC's ROP, with safety performance indicators (PIs) currently being used to disclose safety information about nuclear facilities to the public. On the other hand, the security sector has developed a security ROP framework and is progressing in the development of security PIs.

In this study, we assessed the suitability of the developed security PIs from a regulatory perspective using three evaluation criteria. First, we examined whether nuclear operators can independently collect these indicators. Second, we evaluated whether the indicators sufficiently reflect security performance. Third, we assessed whether these indicators can be used to justify easing regulatory implementation. This study confirms the suitability of the developed indicators and suggests improvements for identified deficiencies.

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Keywords: Safety and security interface, Risk-informed performance-based regulation, Performance indicator, Nuclear security

Nuclear Security Regulatory Implications of IAEA SMR NHSI

Seong Youn Jo* and Eunji Chang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jerry@kinac.re.kr

The role of nuclear power generation is becoming increasingly important as a means of reducing carbon emissions to counter global warming. SMRs, which are being developed worldwide, are expected to drastically reduce the risk of severe accidents in existing LWRs and become a clean energy source without carbon dioxide emissions. However, differences in regulatory regime in each country are expected to be barriers to international deployment of SMRs. Accordingly, the IAEA proposed the Nuclear Harmonization and Standards Initiative as a measure to resolve regulatory obstacles that may arise due to differences in the design characteristics, regulations, and applicable industrial standards for SMRs. The IAEA NHSI began its activities in September 2023 and consists of two tracks: regulators and industry. The regulator track operated three working groups: 1) information sharing among regulators, 2) international pre-licensing review, and 3) leveraging of other regulatory reviews, and is scheduled to end Phase I activities in 2024. The NHSI working group mainly discussed regulatory procedures and collaborative measures for advanced reactors from a technology-neutral perspective rather than a specific reactor type. The discussions included information that needs to be shared between regulators during SMR pre-licensing and licensing reviews and methods for sharing, procedures for implementing actual multinational pre-licensing joint reviews, and ways to leverage regulatory review results from other agencies, taking into account differences in experience and capabilities between regulatory agencies. The results of Phase I suggest several tasks to be addressed, such as the scope of information sharing between member states, the need for separate cooperation arrangements, and whether to include reviews for specific reactor types or specific designs. In the second phase, a plan is in place to formulate multinational pre-licensing joint reviews of specific design, build a repository for information sharing, and develop regulatory cooperation toolkits. In the first phase, discussion topics were addressed mainly in the view of safety regulation with little interest on nuclear security. In this regard, the formation of a new working group for security is being considered in the second phase. This paper summarizes the key outcomes of NHSI's Phase I activities, suggests implications for future SMR regulation and considerations for Phase II from a nuclear security perspective.

Keywords: NHSI, Nuclear security, SMR, Pre-licensing review, Regulatory information sharing

Study on the Development of Regulatory Guidelines for the Physical Protection System (Intrusion Prevention System) of Small Modular Reactor

So Eun Shin*, Ha Neul Na, Youngsuk Bang, and Heung Gyu Park

Future and Challenge Tech.

*shin0811@fnctech.com

Small Modular Reactors (SMR) under development worldwide differ from existing large nuclear power plants in several ways, including installation location and size, nuclear fuel transportation and storage, and spent nuclear fuel management. Consequently, the design of Physical Protection System (PPS) for the SMR also varies. The physical protection is under the category of the nuclear security, which also includes cybersecurity. In Korea, the Korea Institute of Nuclear Nonproliferation and Control (KINAC) oversees the nuclear security regulations.

An innovative Small Modular Reactor (iSMR) is currently developing in Korea. Thus, new physical security requirements suitable for the unique characteristics of SMR and the domestic environment must be established to protect against the new threats specific to iSMR. It is essential to ensure that the level of nuclear security for SMR is not degraded compared to large nuclear power plants because SMR's enhanced safety and security features will make the accomplishment of radiological sabotage.

The PPS for the existing nuclear power plants has been regulated according to the regulation standards of KINAC. However, these standards do not yet consider the design characteristics of PPS for advanced nuclear reactor technologies, such as iSMR. Therefore, the regulation guideline for SMR PPS must be developed by revising and enhancing the existing regulation system used for existing large nuclear power plants, through a gap analysis between SMR and existing large nuclear power plants.

Meanwhile, one of the crucial components of PPS is the intrusion prevention system, which acts as a physical barrier (e.g., fences) to deter or prevent unauthorized access. This system serves as the first line of defense against external threats, delaying intrusion and providing security staff with response time. In this study, technology trends in the design of intrusion prevention systems for SMR PPS, both domestically and internationally, are reviewed.

In particular, the U.S. Nuclear Regulatory Commission (NRC) is revising its regulations to offer alternatives to PPS for advanced nuclear reactors. Therefore, it also aims to propose the direction for development of domestic regulatory guidelines for intrusion prevention system of SMR, based on a review of international regulatory practices, with focus on the case of the U.S.

Keywords: Small Modular Reactor (SMR), Physical Protection System (PPS), Intrusion prevention system

Development of a Guideline to Examine International Transport Security Plans of Nuclear Material

Jae-woong Tae

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

tjwjtjw@kinac.re.kr

In Korea, an international transport security plan (TSP) must be approved in advance by the Nuclear Safety and Security Commission (NSSC) in order to transport nuclear materials requiring physical protection internationally according to the revision of the act on physical protection and radiological emergency in 2021. The aim of this study is to identify key considerations for review of a TSP and to develop a guideline to help nuclear licensees or carriers to examine their TSPs. We examined actual cases of international transportation of nuclear materials and the related TSPs. Since most nuclear material transported internationally is low-enriched uranium for use in light water reactors, analyzed cases are limited to maritime transport of low-enriched uranium in this study. In conclusion, key considerations such as transport routes, means of transport and their flag state, transport containers, and emergency plans were derived and the guideline was developed consisting of approaches to deal with key considerations. The guideline would probably be useful either as a reference work for nuclear licensees or carriers who intend to transport nuclear material internationally. It is also expected that the TSP review period will be shortened .

Keywords: Physical protection, International transport, Nuclear material, Nuclear security, Transport security plan, TSP

Study on UK Security Regulation for ROK Security Regulation Gap Analysis

Subong Lee* and Eunji Chang

Korea Institute of Nuclear Nonproliferation And Control (KINAC)

*sblee@kinac.re.kr

Currently, the Republic of Korea (ROK) is in the process of developing an innovative SMR (i-SMR) capable of producing up to 170 MW(e). Due to the different design aspects of i-SMR, the applicability to existing security requirements should be analyzed, and modifications may be required. Therefore, security regulatory strategy for i-SMR are necessary, such as the documentation, requirement, methodology based on safety and security integration.

This study aims to analyze the security regulatory gaps based on the security regulation case by UK Office for Nuclear Regulation (ONR) in order to develop security regulatory documentation and to utilize them as references. The General Design Assessment (GDA) is a process for the safety and security assessment of new nuclear power plants in the UK. This process will be applied where the ONR is asked to assess a proposed design in advance. By applying the GDA process, regulatory considerations can be addressed from the early design stages of new nuclear power plants, thereby reducing regulatory evaluation time and resolving potential regulatory uncertainties that site operators might face in the future. The GDA process consists of three steps in total, with the ONR defining the evaluation objectives and security-related requirements for each step. To conduct the GDA, the ONR evaluates the suitability of applicants' safety-security cases across the 22 technical assessment topics, with security being one of the required items.

The ONR utilizes Security Assessment Principles (SyAPs) and supporting Technical Assessment Guides (TAGs) to guide security regulatory judgements when conducting security regulatory assessments of the applicant's security submission. The SyAPs provide essential foundations for introducing outcome-focused regulation across all security disciplines: physical, personnel, transportation, cyber security, and information assurance. The SyAPs are supported by TAGs to further assist regulator's decision-making and providing guidance for each technical security aspect.

According to the UK ONR licensing framework, the "Security by Design" concept is applied by considering security as a technical assessment topic during the GDA process. This approach ensures robust security by integrating physical and cyber security into the reactor design stage. The dualization of laws regarding nuclear safety and security in the ROK makes it challenging to apply "Security by Design" concept. Therefore, as evidenced in the UK ONR licensing framework, it is essential to ensure that "Security by Design" concept is integrated from the design stage of SMR within the ROK licensing framework.

Keywords: SMRs, Security, Regulation, Documentation, UK

Cybersecurity for Critical Digital Assets in the Vital Area of Nuclear Facilities

Seungmin Kim* and Hyunjoo Lee

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*smkim90@kinac.re.kr

Nuclear licensees should identify Critical Digital Assets (CDAs) and implement appropriate protective measures to protect nuclear facilities from potential cyberattacks. CDAs are identified based on the performance of Safety, Security, and Emergency Preparedness (SSEP) functions, the provision of access paths, and the support and protection of critical systems and CDAs. CDAs comprise a significant portion of digital assets, and it is impractical to apply the same cybersecurity measures to all identified CDAs. To address this issue, assets whose functional incapacitation due to cyberattacks could lead to core damage are defined as Vital Digital Assets (VDAs). However, additional verification of the Probabilistic Safety Assessment (PSA) model used identifying VDA is necessary before applying cybersecurity regulations. This paper proposes an enhanced cybersecurity approach by leveraging the Vital Area (VA) concept, which is already applied in the physical protection regulations for nuclear facilities.

The PSA model is utilized to identify VAs, assuming the loss of compartment functionality due to physical attacks. However, the identification of VAs does not adequately account for the risk of cyberattacks on CDAs. Even if physical attacks are effectively mitigated, the possibility of cyberattacks compromising VA functionality remains. Therefore, CDAs located within VAs should be subjected to enhanced protection measures. The proposed approach to strengthening protective measures for CDAs involves augmenting cybersecurity controls and applying a defense-in-depth strategy. Cybersecurity controls include technical, operational, and managerial security measures. Enhancing cybersecurity controls aims to reduce reliance on alternative measures—such as operational or managerial controls—when technical controls are not feasible. For example, KINAC/RS-015 outlines 101 cybersecurity controls for CDAs. However, using alternative measures in place of technical controls could result in fewer than 101 controls being applied. Consequently, it is essential to minimize the use of alternative measures for CDAs within VAs. In terms of operational and managerial controls, one potential enhancement involves increasing the frequency of regular inspections. As part of the defense-in-depth strategy, CDAs within VAs should be classified at the highest security level. Furthermore, when such CDAs communicate with CDAs outside the VA, boundary protection systems should be implemented to ensure additional layers of security.

This paper proposes enhanced cybersecurity controls for CDAs located within VAs. The first approach is to strengthen the application of technical controls (minimizing the use of alternative measures) and to shorten the interval between periodic audits. The second approach is to classify CDAs within VAs at the highest level of the defense-in-depth strategy, and to implement boundary protection systems when these assets communicate with CDAs outside the VA.

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Keywords: Cyber security, Nuclear facility, Critical digital asset, Vital area

A Study on Improving Domestic Nuclear Cybersecurity Inspector and Inspector Qualification Program Through NRC Case Analysis

Song Lee*, Taehee Kim, and Subong Lee

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*sssong@kinac.re.kr

Given the increasing recognition of cybersecurity as a critical element for the safe operation of nuclear facilities, there is a need to adopt a more structured approach to enhance the capabilities of cybersecurity inspectors. This study aims to analyze the cybersecurity inspector and inspector qualification program of the U.S. Nuclear Regulatory Commission (NRC) and compare it with the Republic of Korea's (ROK) nuclear regulatory inspector training program to propose an improvement plan.

The NRC's inspector qualification program, as outlined in Inspection Manual Chapter (IMC) 1245, is designed to systematically enhance inspectors' competencies in legal, technical, regulatory, and interpersonal domains. The program consists of three stages: 1) Basic level training, 2) Proficiency level training, and 3) Final qualification activities. Notably, proficiency level training is divided into general and technical proficiency training, with technical proficiency further segmented into 16 specialized areas, including Operations, Engineering, Security, and Cybersecurity.

The NRC's cybersecurity inspector qualification program includes comprehensive technical proficiency training through Individual Study Activities (ISA) and On-the-Job Training (OJT), covering cybersecurity regulations, Cyber Security Plan (CSP) implementation procedures, Critical Digital Assets (CDA) and Critical Systems (CS) identification and the cybersecurity Significance Determination Process (SDP). This training process strengthens the regulatory oversight of nuclear facility security and enhances the credibility of the regulatory body. If Korean regulatory bodies adopt a similar model, it could significantly improve cybersecurity regulation within the nuclear industry and strengthen national nuclear safety and security through a more systematic and efficient regulatory approach.

In contrast, the ROK's nuclear regulatory inspector training governed by the internal training regulations of Korea Institute of Nuclear Safety (KINS), is divided into 1) Basic training and 2) Field-specific training courses. These fields include Facility Management, Radiation Management, Quality Assurance, and Radiological Emergency Preparedness. The certification areas for nuclear inspectors include Facility Management, Radiation Management, Quality Assurance, International Regulatory Materials, Radiological Emergency Preparedness, and Physical Protection. Compared to the NRC program, the ROK's training is relatively more straightforward and there appears to be a need for training to enhance the technical expertise.

Furthermore, the ROK's regulations on nuclear non-proliferation and nuclear security are governed by the Act on Physical Protection and Radiological Emergency (APPRE), and the nuclear safety is governed by Nuclear Safety Act. However, given the interconnection of many systems within nuclear facilities, there is a growing emphasis on Safety-Security Interface (SSI). In the case of the NRC, all inspectors, including those specializing in cybersecurity and security, must complete basic level training to build foundational competencies related to reactor safety. This indicates that SSI has been considered in inspector training. Therefore, the ROK should also consider implementing an inspector qualification program that integrates nuclear safety and security requirements and prepares for SSI. This program would acknowledge system interdependencies and ensure that safety and security measures are appropriately integrated, enhancing overall system reliability.

Keywords: NRC inspector qualification program, Nuclear security, Cybersecurity, Technical proficiency level training, Cybersecurity inspector, Safety-Security Interface (SSI)

Development and Evaluation of a Staged Review System for Cybersecurity Regulations in Nuclear Facilities in the Republic of Korea

Hyunjoo Lee* and Eunji Chang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*sn220865@kinac.re.kr

The increasing prevalence of digital equipment in nuclear power plants and low- and intermediate-level radioactive waste management facilities has led to an expansion in the use of digital technology and the establishment of cybersecurity program. In the Republic of Korea, nuclear operators are required to establish their facility's cybersecurity program in accordance with the "Act on Physical Protection and Radiological Emergency (APPRE)" and to specify the requirements of these programs in their own Cybersecurity Plan (CSP). This CSP must be submitted to the Nuclear Safety and Security Commission (NSSC) for approval prior to the loading of nuclear fuel. The Korea Institute of Nuclear Nonproliferation and Control (KINAC), the delegated agency for nuclear security regulation on the nuclear facility, is responsible for reviewing the adequacy of this CSP and requesting approval from the NSSC.

The review process for nuclear facilities is conducted in two main steps: Construction Permit phase and the Operational License phase. During these stages, the KINAC reviews the cybersecurity of the CSP to ensure that the CSP met the requirements on the APPRE and the regulatory standards of KINAC. However, with advancements the technologies of off-the-shelf design for nuclear facilities, such as Small Modular Reactors (SMRs), a Standard Design Approvals stage has been introduced prior to the construction permit phase. This new stage includes the assessment of cybersecurity aspects, contributing to a more comprehensive review system.

This study analyzes the requirements for cybersecurity regulation reviews for domestic nuclear facilities, including radioactive waste management facilities, and identifies the aspects that can be verified at each stage of the Construction Permit and Operational License. Based on this analysis, a review system detailing the staged review items is proposed. Additionally, an advanced staged review system is developed to include cybersecurity review items that can be verified during the Standard Design Approval phase, reflecting advancements in standard design technologies. The proposed staged review system was pilot-tested in the actual review process for nuclear facilities to evaluate its applicability in assessing the adequacy of nuclear facility CSP.

Keywords: Cyber security plan, Review, Nuclear facility, Nuclear security

Analysis of Cybersecurity Regulatory Requirements for Czech Nuclear Facilities

Junghyun Moon*, Taehee Kim, and Subong Lee

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*moon_aver@naver.com

Following South Korea's selection as the top candidate for constructing a new nuclear power plant in the Czech Republic, there has been growing interest in the Czech nuclear industry. The Czech Republic, which presently runs six nuclear power plants, plans to raise the proportion of nuclear energy in its electricity production from the present 36.6% to between 46% and 58% by 2040, as outlined in its State Energy Policy (SEP).

The Czech Republic, as a ratifying country of the Convention on the Physical Protection of Nuclear Material (CPPNM) and its Amendment (A/CPPNM), fulfills its international obligations by regulating nuclear material through the State Office for Nuclear Safety (SÚJB). The SÚJB, established under the Czech Atomic Act, defines key rules for implementing physical protection as mandated by the Atomic Act, with cybersecurity regulatory requirements specified in section 163, paragraphs a & b of No. 263/2016.

The Decree on Security of Nuclear Facilities and Nuclear Material (No. 361/261) is based on IAEA Nuclear Security Series (NSS) No. 17, which addresses computer security for nuclear facilities. It includes cybersecurity provisions in sections 18, 19, and 28 to safeguard the computer systems and data of nuclear facilities. Section 28 covers the content of documentation for licensed practices in the area of security, including the physical protection assurance plan, which incorporates a Computer Security Plan (CS plan) as specified in par. 3 l. e) p. 3. The CS plan, as the implementation of policy by the security organization, mandates measures for the protection of nuclear facility safety. Section 19 specifies measures for the security of computer systems, such as the prevention of misuse, assessment of security levels, and designation of personnel with the necessary expertise, while Section 18 outlines communication and physical access control measures for the protection of the technical system of physical protection and its data. In addition, the SÚJB ensures that I&C system inspectors conduct periodic cybersecurity assessments, including document management, log reviews, and configuration management inspections, both annually and before start-up, as part of the Safety Assessment of Licensee Performance (SALP) to evaluate cybersecurity.

Safety regulations for nuclear power plants go beyond a simple legal framework and play an important role in strengthening the safety and security of society. In particular, cybersecurity is a critical regulatory factor in the modern nuclear industry, and this paper focused on analyzing the cybersecurity requirements of the Czech nuclear industry. This analysis suggests that these findings can be leveraged to fulfill the cybersecurity regulatory requirements in the collaborative process for upcoming nuclear power plant initiatives.

Keywords: Czechia, Nuclear cybersecurity, Regulatory, Nuclear law

2분과

사용후핵연료 처분전관리 (ORAL)

Graphite-Boron Composites for Innovative Canister Material Used in Spent Nuclear Fuel Dry Casks

Hongin Kim^{1,2}, Sunghwan Yeo^{2,*}, and Young-Kook Lee¹

¹Yonsei University

²Korea Atomic Energy Research Institute (KAERI)

*sy eo@kaeri.re.kr

The radiation shielding and heat dissipation properties of dry cask canister are critical factors for the safety of spent nuclear fuel storage and workers. Stainless steel is conventionally utilized canister material mainly due to its excellent mechanical property. While it may be suitable as a structural material, it is not effective for neutron shielding and heat dissipation. Graphite and boron are selected as innovative composite material can be used for canister instead of stainless steel. It is well known that graphite has superior thermal conductivity and boron obtains excellent neutron shielding capabilities. In this study, graphite-boron composites are fabricated through a powder sintering process including mixing, pressing, and heat treatments from natural and synthetic graphite, boron carbide (B_4C), and boron nitride (BN_{HCP}) powders. The mechanical and thermal properties of fabricated composites were analyzed using optical and scanning electron microscopes, Vickers hardness tester, and laser flash instrument. particle size and compaction pressure were varied to investigate optimal conditions. In addition, to evaluate the neutron shielding performance of graphite-boron composites compared to conventional SUS304, neutron penetration tests were performed using the Monte Carlo simulation program Geant 4 code. Fabricated graphite-boron composites exhibited significantly improved thermal conductivity when compared to SUS304 mainly due to the graphite. In the horizontal direction, the thermal conductivity was similar to that of SUS304, while in the vertical direction, it was 2.4 to 3.8 times higher, depending on the temperature. Enhanced neutron shielding property at thermal neutron energy spectra of graphite-boron composites was attributed to included boron content. The neutron absorption cross-section was 29.2 times higher than SUS304.

Keywords: Dry cask storage, Canister, Graphite-boron composites, Powder metallurgy

Development of Image Reconstruction Algorithm Based on Prior-aware Information and Machine Learning to Inspect Partial-defect Within PWR-type Spent Nuclear Fuel

Hyung-Joo Choi¹, Jaewon Jeong², Hakjae Lee², Sung-Woo Kwak³, and Chul Hee Min^{1,*}

¹Yonsei University

²ARALE Laboratory Co., Ltd.

³Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*chmin@yonsei.ac.kr

Spent nuclear fuel (SNF) is a byproduct of nuclear power generation. It is classified to be high-level waste due to its radioactivity and heat capacity, therefore, loss and theft of SNF could lead to unwanted harmful radiation release to biosphere. For this reason, the accurate and effective inspection method has been required for the nuclear safety. In our previous study, we developed the gamma emission tomography instrument named Yonsei Single-photon Emission Computed Tomography (YSECT) to detect the partial-defect within the SNF. However, the YSECT showed the low-accuracy for central region of SNF due to the high-density of nuclear fuel rod. The aim of this study is to develop the image reconstruction algorithm to improve the partial-defect detection accuracy for the entire region of SNF.

The developed image reconstruction algorithm follows three steps: Monte Carlo (MC)-based big data build process, iterative reconstruction process with prior-aware information, and machine learning-based denoising process. At first, to build the big data of machine learning model, the sinograms were obtained using MC simulation. Secondly, the iterative reconstruction algorithm was developed based on the prior-aware information to improve the pixel intensity for the central region. Finally, the machine learning-based denoising technique was developed to improve the image quality. In order to evaluate the performance of developed image reconstruction algorithm, the image quality was compared between the Filtered Back-projection (FBP), iterative reconstruction algorithm, and developed algorithm. In addition, the pattern classification accuracy was analyzed for the three patterns.

On the basis of MC simulation and developed reconstruction algorithm, total number of 6,000 tomographic images were obtained for the training and test of machine learning model. To validate the performance of prior-aware information-based iterative reconstruction, the 6 images were selected taking into account of partial-defect distribution ratio. Based on the comparison between reconstruction algorithms, the tomographic image quality was enhanced for all patterns. However, the noise was occurred for central region due to the over-correction. For the improvement of image quality, the machine learning-based denoising technique was employed. On the basis of the validation results of pattern classification accuracy, the pattern classification accuracy was calculated to be about 100%. We expected that the developed reconstruction algorithm could improve the image quality for entire region of SNF.

In current study, the reconstruction algorithm based on the prior-aware information and machine learning model was proposed and developed to enhance the partial-defect detection accuracy for the entire region of SNF. In future, the performance of developed reconstruction algorithm will be experimentally evaluated using the mock-up of SNF and pool.

Acknowledgements

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Keywords: Safeguards, Spent nuclear fuel, Partial-defect, Gamma emission tomography, Tomographic imaging

Results of Korean Transportation Tests Under Normal Conditions of Transport

JaeHoon Lim* and Woo-seok Choi

Korea Atomic Energy Research Institute (KAERI)

*jhl85@kaeri.re.kr

In 2017, an ENSA/DOE multi-modal transportation test (MMTT) was conducted using the ENSA ENUN 32P transportation package which is without a canister and possesses an integrated internal structure. The strain and acceleration data were obtained for three different modes of transport. In Korea, the development of evaluation technology for vibration and shock loads transmitted to spent nuclear fuel and structural integrity of spent nuclear fuel under normal conditions of transport was performed by the present authors. The road and sea transportation tests were performed using the test model of KORAD21 cask, which features a canister and its internal structures divided into a disk and baskets. Additionally, the shake table tests and separate rolling test were also conducted.

By integrating the findings from both MMTT and the Korean road and sea transportation tests, it becomes evident that the tests were carried out under diverse conservative conditions across three different modes of transportation: railway, road, and sea but acceleration and strain that could threaten the integrity of SNF were not measured. In other words, the Korean road and sea transportation tests complemented the MMTT, and the stresses fuel rods experience due to vibration and shock during normal transportation are far below the yield and fatigue limits for cladding. Despite using casks with different characteristics such as the presence or absence of canister and separate basket structure, the measured strain values were very low in both cases. Therefore, it can be concluded that the structural integrity of spent nuclear fuel is maintained under normal transport conditions.

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Keywords: Normal conditions of transport, Spent nuclear fuel, Shock and vibration, Structural integrity

Development of Verification Tools for Ensuring Spent Nuclear Fuel Retrievalability Based on the IAEA Approach

Ji-Young Kim*, Sun-Hee Choi, and Seong-Ki Lee

KEPCO Nuclear Fuel (KEPCO NF)

*jiyoung@knfc.co.kr

The retrievability of Spent Nuclear Fuel (SNF) is a critical factor in its long-term management, especially under dry storage conditions as defined by the International Atomic Energy Agency (IAEA). It refers to the ease with which bare fuel bundles, including rods, can be removed from a cask or canister. The primary goal is to optimize this process to ensure smooth retrieval and subsequent handling without issues arising from deterioration of the fuel rod cladding, structural integrity of the assembly, or difficulties related to cask handling. To achieve this, the desired conditions at the time of retrieval include no leaks in the fuel rod cladding and no failure or distortion of the fuel assembly structure. Key factors influencing retrievability are the integrity of the fuel cladding and assembly structure, and the robustness of the cask's internal configuration. Proper management of these factors provides reasonable assurance of routine retrieval and handling, regardless of decisions on the final disposition of SNF, such as direct disposal or reprocessing.

Concerns about retrievability grow with increased storage durations, especially for extended or initially undefined periods. Technical requirements for retrievability, therefore, focus on the long-term containment of radioactivity by imposing design limits on cask thermal loading and cladding temperature to maintain fuel rod cladding integrity. This approach reduces the risk of fuel cladding and assembly structural failures, thereby improving retrievability in the future.

To demonstrate compliance with the IAEA's definition of retrievability for national nuclear fuel management systems, it is essential to develop reliable solutions, such as integrity evaluation codes and equipment. This study introduces a comprehensive set of tools, including computational codes, fuel degradation analysis, and mechanical Finite Element (FE) models. These tools are designed to assess storage conditions, predict potential structural failures, and ensure the integrity of fuel assemblies. Additionally, the development of Vacuum Canister Sipping (VCS) equipment provides a reliable method for inspecting and detecting potential fuel damage. Developing these tools and technologies domestically is critical for maintaining the physical integrity of SNF, reducing reliance on foreign technology, and addressing local requirements.

The approach involves establishing retrieval solutions through the development of integrity assessment codes/models and inspection equipment. These items have undergone rigorous verification and validation processes, as well as comprehensive testing under the advanced quality assurance system of Kepco Nuclear Fuel (KNF).

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Keywords: Retrievability, Spent nuclear fuel, Integrity evaluation code/model, Integrity inspection equipment

Effects of Surface Temperature on Chloride-induced Stress Corrosion Cracking Behavior of Stainless Steel Welds for Dry Storage Canister

Seunghyun Kim*, Gidong Kim, and Sang-Woo Song

Korea Institute of Material Science (KIMS)

*skims@kims.re.kr

This study investigates the potential for Chloride-induced Stress Corrosion Cracking (CISCC) in spent nuclear fuel dry storage containers. The research specifically considers the surface temperature of the dry storage containers, which has not been addressed in previous experiments.

In this study, various materials considered as candidates for dry storage containers, such as 304 L-ER308 L, 316 L-ER316 L, and Duplex Stainless Steel (DSS), were evaluated. Using a cyclic corrosion chamber and a custom-made heater, the effect of heating the surface of the dry storage containers to temperatures of 75°C and 100°C, while maintaining an ambient environment of 60°C, was assessed. According to the evaluation results, an increase in temperature within various types of welds did not lead to the deliquescence of salt, resulting in a tendency for reduced occurrences of both CISCC and pitting corrosion. Therefore, based on the results of this study, it can be concluded that dry storage containers can maintain structural integrity well when initially exposed to high temperatures. However, CISCC may occur after the nuclear fuel has cooled.

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Keywords: Dry storage canister, Chloride-induced stress corrosion cracking, Stainless steel welds

Estimation of Fracture Parameters of Spent Nuclear Fuel Cladding With Reoriented Hydride Under Pinch Load

Seyeon Kim and Sanghoon Lee*

Keimyung University

*shlee1222@kmu.ac.kr

Ensuring cladding integrity is critical to the safe and economic management of spent nuclear fuel (SNF). In horizontal drop accidents, the most damaging cask drop orientation, fuel rods are subjected to bending and pinch loads, which can severely affect the structural integrity of the SNF rod. High burnup SNF cladding may contain reoriented hydrides and mechanical defects, with the radial hydride fraction (RHF) playing a key parameter in determining fracture resistance. High RHF values lead to a significant reduction in resistance to Mode-3 failure under pinch load. The Ring Compression Test (RCT) is commonly used to evaluate the mechanical behavior of cladding under these conditions.

This study presents a method for estimating fracture parameters of cladding with reoriented hydrides and systematically evaluates the fracture resistance of cladding under pinch load. A finite element model was developed using a ductile damage model based on continuum damage mechanics to simulate cracking during the RCT, accounting for interactions between the Zr matrix, hydrides, and the interface. The fracture parameters of the ductile damage model, particularly the equivalent strain values under various stress triaxiality conditions and the fracture energy values, were selected to accurately model damage initiation and evolution. However, challenges remain in determining accurate fracture parameters due to limited experimental data. To address this, fracture parameters are estimated using a Kriging metamodel and optimization techniques. The developed RCT simulation model using pixel-based finite elements cannot be directly used for optimization of fracture parameter calibration due to its high computational cost. Therefore, a Kriging metamodel was developed to approximate the fracture behavior of cladding in the fracture parameter space and enable efficient parameter estimation. A Design of Experiments (DoE) approach was used to generate a comprehensive dataset, and an Analysis of Variance (ANOVA) was performed to evaluate parameter effects. The Kriging metamodel was constructed to account for uncertainties due to variations in hydride morphology and RHF in the RCT specimen. This metamodel was then used to optimize and estimate fracture parameters by fitting RCT force-displacement curves. Both the mean and variance of the estimation were considered in a multi-objective optimization framework. The estimated parameters enable the prediction of crack initiation under pinch load for uncertain RHF distributions within the same specimen. It is expected that the developed simulation method, together with the parameter estimation framework, will provide a systematic way to quantify the fracture resistance of SNF cladding under pinch load while accounting for uncertainties.

Acknowledgements

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Keywords: Spent nuclear fuel, Hydride reorientation, Pinch load, Crack propagation, Parameter estimation, Metamodeling

Development and Application of ^{235}U and Plutonium Mass Estimation Formulas to Spent Nuclear Fuel Assemblies Using Differential Die-Away (DDA) Analysis Database

Sehwan Seol and Ser Gi Hong*

Hanyang University

*hongsergi@hanyang.ac.kr

South Korea is facing challenges in spent nuclear fuel management, with storage facilities at nuclear power plants approaching their capacity limits in the coming years. The current inventory of spent fuel is nearing saturation levels, highlighting the need for effective characterization and management strategies for nuclear materials, for the safe transportation and disposal of spent nuclear fuel.

The Differential Die-Away (DDA) method involves generating neutrons using a pulsed neutron source and measuring the neutron signal decay time, known as die-away time (τ). This decay time can vary depending on the structure and composition of the target material. When applied to spent nuclear fuel assembly with identical structures, τ changes solely due to the fuel's material composition, which is influenced by characteristics such as initial uranium enrichment, burnup, and cooling time. Consequently, the DDA method can be used to estimate spent fuel characteristics.

In this study, we performed burnup calculations for Westinghouse 17×17 fuel assemblies using the SCALE 6.2 code, generating data for 306 types of spent fuel with varying initial uranium enrichment, burnup, and cooling times. We then modeled the DDA structure using the MCNP 6.2 code to simulate neutron signals and obtain DDA analysis signals: die-away time (τ) and neutron total count (S). This data was compiled into a database, and equations were developed to estimate ^{235}U and plutonium masses in spent fuel using the analyzed τ , S, and cooling time.

When applied to spent fuel assemblies in the database, the developed equations yielded a Mean Absolute Percentage Error (MAPE) of 2.24% for ^{235}U mass estimation and 0.60% for plutonium mass estimation. Additionally, we generated 50 random spent fuel assembly datasets outside the database, assuming unknown spent fuel assemblies with known cooling times, and applied the developed equations. For these unknown assemblies, the ^{235}U mass estimation resulted in a MAPE of 2.29%, with the largest error within 8%. The plutonium mass estimation showed a MAPE of 0.66%, with the largest error within 2.5%, demonstrating high accuracy.

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Keywords: Differential die-away method, Spent nuclear fuel, Nondestructive assay

Proposal for Improving the Method of Estimating Key Design and Operational Parameters of Spent Fuel Using Data on Emitted Gamma-Rays

Sang Hyeok Lee and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Accurate characterization of spent nuclear fuel (SNF) is crucial for ensuring safety and regulatory compliance. This study introduces an enhanced formula-based method for estimating key parameters—burnup, initial enrichment, and cooling time—by analyzing gamma-emitting radionuclides within the fuel. Custom-developed equations were used to establish precise relationships between these radionuclides and the fuel’s operational history. This approach allows for accurate parameter estimation without requiring direct physical testing.

The methodology was validated through extensive simulations on various SNF assemblies under a range of operational conditions. These simulations confirmed that the approach enhances the accuracy of parameter estimation. The results showed that using custom-developed functions significantly improves the estimation of initial enrichment, a parameter traditionally challenging to determine with high precision.

This method provides a reliable, non-destructive assessment tool that is particularly valuable for nuclear facilities in verifying that SNF assemblies meet safety and regulatory criteria. The ability to utilize limited gamma-ray source data also makes this approach adaptable to various reactor types, supporting the development of more accurate and efficient models for SNF management.

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Keywords: Spent nuclear fuel, Non-destructive analysis, Safety, Safeguard

2분과

사용후핵연료 처분전관리 (POSTER)

Current Status of Nuclear Fuel Fabrication and Management Technology Development for Chloride Molten Salt Reactors

Chang Hwa Lee*, Dalsung Yoon, Ju Ho Lee, Seok Min Hong, Yung-Zun Cho, Eun-Young Choi, Taeho Kim,
and Do Kyu Kang

Korea Atomic Energy Research Institute (KAERI)

*chwalee@kaeri.re.kr

Molten salt reactors (MSRs) are advanced reactors that use high-temperature molten salts as both nuclear fuel and coolant, offering an alternative to the solid fuels used in existing light water reactors. Operating at temperatures of approximately 500-700°C without pressurization, MSRs provide exceptionally high thermal efficiency and superior safety, inherently reducing the risk of severe accidents such as meltdowns due to coolant loss. Historically, MSRs have been validated with fluoride-based liquid nuclear fuels through the construction of ARE (Aircraft Reactor Experiment) and MSRE (Molten Salt Reactor Experiment) at ORNL in the 1950s and 1960s. Unlike fluoride-based MSRs, which are suited for thermal neutron spectra, chloride-based molten salt reactors offer high actinide solubility, enabling long-term operation with less need for active online reprocessing. This capability has led to growing research interest in the US and Europe, with development efforts aimed at deploying these reactors by the early 2030s. In Korea, the KARPA project has been developing a chloride-based molten salt reactor for ship propulsion since 2023. This paper reviews the current status of technology development for manufacturing UCl₃-containing initial liquid nuclear fuel and managing fission products generated during MSR operation, evaluating corrosion characteristics of structural materials, and related technologies such as He bubbling, off-gas treatment, and flow loop systems.

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Keywords: Molten Salt Reactors (MSR), Uranium chlorination, Off-gas treatment, Flow loop, He bubbling

The Influence of Solid Solution Hydrogen and Precipitated Hydride on the Creep Behavior of Zircaloy-4

Ho-a Kim, Sungjun Choi, and Sangtae Kim*

Hanyang University

*sangtae@hanyang.ac.kr

Zircaloy-4 is an essential material used in the cladding of nuclear fuel rods, prized for its exceptional strength and outstanding corrosion resistance. However, it can experience deformation over time through a process known as creep, which presents significant safety risks in nuclear reactors. Additionally, the absorption of hydrogen during reactor operation can further compromise its properties and adversely impact its creep behavior.

Previous research suggests a trend in which hydrogen concentration correlates unidirectionally with creep rates, either increasing or decreasing as the concentration rises. This trend is also observed in EPRI's creep model, EDF-CEA Model-3. However, recent literature indicates that creep behavior may vary depending on the state of hydrogen present. Therefore, it has become evident that creep behavior can be influenced not only by hydrogen concentration but also by the state of hydrogen, whether it is in a solid solution state or precipitated as hydrides.

Our study aimed to compare creep behavior in specimens with hydrogen concentrations below and above solubility limits. We fabricated recrystallized annealed (RXA) Zircaloy-4 plate and tube specimens with varying hydrogen concentrations and conducted creep tests. The results revealed that specimens below the solubility limit exhibited decreasing creep rates as hydrogen concentration increased, while those above the limit displayed increasing creep rates. We interpreted these creep test results from a microstructural perspective through electron backscattered diffraction (EBSD) analysis. The EBSD analysis showed that in specimens with hydrogen concentrations below the solubility limit, the proportion of intragranular hydrides precipitated within grains was over 93%, while the proportion of intergranular hydrides at grain boundaries was less than 7%. In contrast, specimens with hydrogen concentrations above the solubility limit exhibited a significant increase in intergranular hydrides, with proportions exceeding 20%.

Based on this microstructural analysis and literature review, we concluded that intragranular hydrides, precipitated within grains, exist in a solid solution state at experimental temperatures, facilitating dislocation movement and accelerating creep progression. Conversely, intergranular hydrides hinder dislocation movement, thereby suppressing creep progression. Currently, EBSD analysis has only been conducted on Zircaloy-4 plates, but we plan to analyze Zircaloy-4 tubes to further understand the creep behavior of spent nuclear fuel cladding.

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Keywords: Creep, Microstructure, Zircaloy-4, Zirconium hydride, Electron backscattered diffraction

Towards Optimal Vacuum Drying Process: Minimizing Drying Time and Residual Water in Spent Nuclear Fuel Canisters for Dry Storage

Ji Hwan Lim^{1,*}, Kyoung-Sik Bang¹, Kyung-Wook Shin², Nam-Hee Lee², and Seung-Hwan Yu¹

¹Korea Atomic Energy Research Institute (KAERI)

²SAE-AN Engineering Corp.

*jlim@kaeri.re.kr

The recent revision of the Industrial Standards Guide by the American Society for Testing and Materials (ASTM) in 2023, addressing the vacuum drying process of spent nuclear fuel, has been noted by leading academia in the field of spent nuclear fuel storage. Academia emphasizes concerns regarding residue accumulation within spent nuclear fuel during dry storage, suggesting awareness of water-related impacts and the necessity for corresponding regulations. The updated guide clarifies the rationale behind drying and offers guidance on verifying the appropriate dryness level. Nevertheless, due to variations in dry storage system environments and the condition of internal spent nuclear fuel, there remains a lack of specific information regarding effective drying methods. Holtec's research report specifies a vacuum drying time limit of 40 hours for spent nuclear fuel multi-purpose canisters with a decay heat of 23-28.74 kW. Additionally, the 2023 study conducted by Oak Ridge National Laboratory on failed nuclear fuel emphasizes the importance of meeting the final drying requirement, which should not exceed saturated water vapor pressure at 3 torr for 30 minutes. Given the diverse external factors such as spent nuclear fuel, canister, and dry storage system environment, standardizing an effective drying method for residual water inside the canister poses a challenge. However, by focusing thermodynamically on the two-phase path of water inside the canister and its movement within the phase-diagram, it's possible to develop an optimal vacuum drying route that can be tailored to each situation, potentially shortening the drying time. This study aims to quickly reach the saturation pressure condition, which varies depending on water temperature and canister internal pressure, and maintain this situation to maximize water evaporation below the specific saturation pressure. This approach is expected to greatly assist Korea's domestic academic community in developing its own optimal vacuum drying method for industrial use, thereby enhancing the safety and reliability of spent nuclear fuel integrity after the dry storage period. In the future, the research team plans to scale up the equipment to the pilot stage canister and vacuum drying equipment stage to evaluate the suitability of this theory in a demonstration scenario.

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Keywords: Vacuum drying, Spent nuclear fuel, Evaporation, Residual water, Heat-transfer

Improvement of Regulations on On-site and Off-site Transport of Domestic Spent Nuclear Fuel

Kiyoung Kim*, Donghee Lee, Beomgyu Kim, and Keongho Noh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kiyoungkim@khnp.co.kr

Nuclear power generation, which has recently attracted worldwide attention, plays an important role in meeting the energy demand of modern society due to the rapid spread of AI as well as electric vehicles, and the safe management and transportation of spent nuclear fuel is one of the key elements of nuclear power plant operation. In particular, Korea and the United States have different approaches to the method of transporting spent nuclear fuel within nuclear power plant sites. This difference has important implications in terms of the safety management of spent nuclear fuel, and is clearly revealed in terms of technical standards, legal regulations, and actual operation. In this paper, we will review the technical standards related to the use of spent nuclear fuel transport containers and transfer containers within nuclear power plant sites in Korea and abroad and suggest improvement measures for them.

In countries operating nuclear power plants, a distinction is made between transfer casks and transport casks for the transportation of spent nuclear fuel. Generally, transfer casks are used for on-site transportation operations within the boundaries of the nuclear plant, carried out under strict safety regulations and controls. On the other hand, transport casks are utilized for off-site transportation, such as on public roads, railways, or through ports, and are therefore regulated under stringent standards, such as those set by the International Atomic Energy Agency (IAEA).

In South Korea, the site boundary for classifying on-site and off-site transportation of spent nuclear fuel is defined at the plant boundary. Accordingly, current regulations apply on-site transportation rules within the plant, and off-site transportation rules for transport between plants (within the same complex). For transportation to dry storage facilities for CANDU spent nuclear fuel, off-site transportation standards are applied.

In contrast, the IAEA's safety standard SSR-6 (Rev. 1) for the transportation of spent nuclear fuel states that "Transport regulations do not apply to the transport of radioactive material that is carried out within an establishment under appropriate safety provisions and does not involve public roads or railways". Similarly, in the United States, 10 CFR 71 specifies that "Transport regulations apply when licensed radioactive materials are transported outside of the site of use or on public roads". Thus, in the United States, when transporting spent nuclear fuel to on-site dry storage facilities, transfer casks may be used in accordance with on-site transportation standards.

The transportation environment within a nuclear power plant site is one where civilian access and vehicle movement can be strictly controlled, allowing for immediate responses to various potential incidents that may occur during transportation. Additionally, the establishment and adherence to operational procedures minimize the likelihood of events during transportation. This is further supported by the extensive experience accumulated from the transportation of spent nuclear fuel since the 1990s. Therefore, as an area for improvement related to the transportation of spent nuclear fuel in South Korea, it is necessary to relax regulations by applying on-site transportation standards to transportation within the nuclear plant site, thereby allowing the use of transfer casks. Specifically, for the efficient operation of spent fuel (SF) dry storage facilities within nuclear plant sites, the site boundary standard that currently distinguishes between on-site and off-site transportation, as stated in the NSSC (Nuclear Safety and Security Commission) notice, should be changed from the plant boundary to the entire nuclear complex. This change would simplify transportation procedures and enhance safety, including reducing radiation exposure to workers.

Keywords: Transfer cask, Transport cask, On-site, Off-site, 10CFR71

Fixing Device for Safely Handling Spent Fuel Assembly and Its Installation/ Removal Tools

Joung Yeul Lee*, Hyeong Koo Kim, Jae Jun Lee, Seong Ki Lee, and Dong Geun Ha

KEPCO Nuclear Fuel (KEPCO NF)

*joungyeul@knfc.co.kr

For the dry interim storage of the spent fuels temporarily stored in the spent fuel pool in the 26 NPPs (nuclear power plants) in Korea, safe handling of the spent fuels should be secured. In 2001, while handling a Westinghouse-type spent fuel assembly at the North Anna NPP in the United States, a top nozzle separation incident occurred. As a result, a safety issue with handling certain types of spent fuel was identified.

The US NRC investigated the cause of the incident and determined the cause was due to degradation of the spent nuclear fuel assembly. As a result, NRC announced in NRC Information Notice 2002-09 that the bulge under the top nozzle among the spent fuels stored for a long time is vulnerable to IGSCC (Inter Granular Stress Corrosion Cracking) which could result in the guide tube and top nozzle separating during lifting. For this reason, fuel using SS-304 sleeves should be handled using special precautions, and should be classified as 'Damaged fuel' according to ISG-1. Accordingly, in the United States, the fuel classified as 'Damaged fuel' can be reclassified as 'Undamaged fuel' by taking reinforcement measures so that it can be handled with a general handling device.

The 'Damaged fuels' currently stored in Korea NPPs are STANDARD fuel assemblies using SS-304 sleeves (supplied by overseas manufacturers), and the quantity is about 1,000 fuel assemblies. To confirm the handling safety of these fuels, KNF developed the Fixing device (called SAFER: Safely Adapting FastenER) for safely handling spent fuels assembly and its installation/removal tools.

Keywords: Spent nuclear fuel, Fixing device, SAFER, Installation/removal tool

Evaluation of Roller Burnishing Treatment on Scaled Canister Model to Mitigate Chloride-Induced Stress Corrosion Cracking

Ki-Hwan Kim*, Sang-Gyu Park, Yun-Young Yang, and Sang-Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*khkim2@kaeri.re.kr

The canister in the spent nuclear fuel storage system serves as a containment barrier for spent fuel and radioactive materials. The canister has multiple welding lines with high residual tensile stresses after welding procedure. The interaction between austenitic stainless steel and chloride environment of sea forms detrimental conditions that cause chloride-induced stress corrosion cracking (CISCC) in the canister. When corrosive pits are created by localized corrosion in seawater atmosphere, CISCC cracks occur and grow. Surface stress improvement, such as roller burnishing, can effectively mitigate the possibility of CISCC on the canister surface. The generation of a compressive stress layer can inhibit the transition to cracking initiation. Removing tensile stress in the canisters can reduce the likelihood of crack initiation. In this study, the feasibility of a roller burnishing method was evaluated to applicable to a 1/3 scaled canister model in order to develop the mitigation technology of CISCC of the canister.

1/3 scaled canister models made were fabricated using gas tungsten arc welding (GTAW) method. A burnishing apparatus was manufactured with 2 sets of burnishing rollers under a canister model. Burnishing conditions were selected on the basis of previous burnishing experimental results of SUS 316 plate-type weldment. Burnishing parameters were adjusted with burnishing load of 100 kgf, burnishing speed of 6.3 m/min, pass No. of 4 turns and feeding rate of 18 μ m/sec. The surface roughness was measured before and after roller burnishing. Surface residual stress was measured by the hole drilling contour method with strain gauges attached before and after roller burnishing.

The surface roughness of 1/3 scaled cylindrical stainless steel canister model after roller burnishing of 4 turns in pass No. showed 1.3 μ m in the base metal and 1.1 μ m in the HAZ parallel to welded line, individually improved by 70% at base metal and 36% with HAZ. GTAW welded canister models had residual compressive stress of yield stress level up to 0.8 mm deep from the surface after burnishing. Roller-burnished canisters with good surface roughness can reduce the number of crack initiation sites. The residual compressive stress might prevent crack initiation by eliminating tensile residual stress in weld zone. The burnished canister with good surface roughness and high residual compressive stress might suppress crack initiation in the CISCC of the weld zone and finally exhibits to excellent CISCC resistance.

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Keywords: Spent fuel dry storage canister, Scaled model, Austenitic stainless steel, Chloride Induced Stress Corrosion Cracking (CISCC), Roller burnishing, Residual stress

Design Considerations for Semi-integral LOCA Test Equipment Used in the PIEF Hot Cell

Myungchul Park*, Hyongmun Kwon, Sunggeun Kim, and Dosik Kim

Korea Atomic Energy Research Institute (KAERI)

*mcpark@kaeri.re.kr

A Loss of Coolant Accident (LOCA) is one of the primary design-basis accidents for nuclear power plants. While regulations concerning the Emergency Core Cooling System (ECCS) performance during a LOCA specify limits for the fuel cladding, evaluating the behavior of nuclear fuel, especially with high burn-up, under various LOCA conditions remains a significant issue. Integral LOCA tests conducted at research reactors have generated many important data by simulating LOCA scenarios with in-pile tests; however, such tests are limited due to the difficulty of finding suitable research reactors and the high costs involved. Although separate effect tests have been performed in the hot cell according to the US NRC regulatory guide, semi-integral LOCA test equipment has also been developed and utilized internationally to simulate the entire LOCA process. Therefore, we aimed to evaluate LOCA-related characteristics of nuclear fuel using the hot cell facility. This paper identifies and reviews the design considerations essential for the establishment of new semi-integral LOCA test equipment within the PIEF hot cell.

Keywords: Hot cell, Loss of coolant accident, Integral LOCA test, Post irradiation examination, Fuel

Direct Dryness Measuring Instrument Suitability Tests for the Dry Storage of Spent Nuclear Fuel

Geon-Hui Lee^{1,*}, Gyung-Sun Chae¹, Tae-hwan Kim¹, Seo-Oh Kim¹, Kyung-Wook Shin¹, Jae-Seok Park¹,
and Seung-Hwan Yu²

¹SAE-AN Engineering Corp.

²Korea Atomic Energy Research Institute (KAERI)

*ghui6@sae-an.co.kr

For long-term dry storage of spent nuclear fuel, the moisture inside the canister that stores the bundles of spent nuclear fuel must be removed. After vacuum drying is completed, the moisture inside the spent nuclear fuel storage system, which is being dry stored for a long time, can be expected to be within the range of 1.0 to 5.5 moles. It is commercially available to use a measuring instrument by an indirect measurement method such as a dew point meter as a means of checking drying suitability after vacuum drying. We conduct an experiment to measure the amount of moisture using an AquaVolt Moisture Analyzer (AMA) measuring instrument that applied the direct measurement method, and compare it with the moisture concentration of the appropriate calibrator (S8000) to confirm the reliability of the measurement results. AMA is an instrument that uses water electrolysis and measures the moisture concentration by directly electrolyzing moisture in the gas sample passing through AMA. It measures the electric current generated in proportion to the moisture content. Sample gas with a known moisture concentration is flowed into the AMA to compare the measured value of the calibrator and the measured value of the AMA. As a result, when the concentration of sample gas is more than 100 ppmv, the measured values is almost identical, and when the concentration of sample gas is less than 100 ppmv, the error is found to be 10 ppmv. Since the background moisture concentration measured after vacuum drying of an empty container without moisture is known to be about 200 to 300 ppmv, it is confirmed that the AMA equipment is suitable for directly measuring residual moisture inside the canister after vacuum drying. AMA, which can measure gas samples under pressure conditions of 1 bar or more, will be used after vacuum drying and helium filling in the canister, and further experiments will be conducted using it as a measuring instrument to confirm drying the adequacy.

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Keywords: Moisture quantification, Residual moisture, Dry storage, Spent nuclear Fuel

Result of Residual Moisture Measurement Tests at 7 Torr Vacuum Pressure for the Moisture Quantification Evaluation Before the Dry Storage of Spent Nuclear Fuel

Geon-Hui Lee^{1,*}, Gyung-Sun Chae¹, Kyung-Wook Shin¹, Hyeong-Seop Jeon¹, Ki-Seong Son¹, Jae-Seok Park¹,
and Seung-Hwan Yu²

¹SAE-AN Engineering Corp.

²Korea Atomic Energy Research Institute (KAERI)

*ghui6@sae-an.co.kr

The NRC's acceptance for the dry storage of spent nuclear fuel is the residual moisture concentration when there is no pressure change for more than 30 minutes at a pressure of 3 Torr inside the dry storage system. When the drying adequacy in the dry storage canister of 7,000 L is satisfied by applying the ideal gas state equation, about 20.3 g of residual moisture may exist in the vapor phase at an atmospheric temperature of 25°C. It is reported that this amount of water vapor has no effect on the degradation of the dry storage system. Most of the drying technologies used for dry storage of spent nuclear fuel are vacuum drying methods, and as a residual moisture measurement method, it is an indirect measurement method using a instrument such as a dew point thermometer. We conduct an experiment to measure the concentration of residual moisture after vacuum drying, and measured the moisture concentration at 7 Torr, the first year's test target, using OPTIDEW, a chilled mirror hygrometer. When the pressure by the vacuum pump reached 7 Torr, the vacuum pump was stopped, and after the pressure was kept constant for more than 30 minutes, helium gas was pressurized to 1.5 bar, and a gas sample was collected and measured. As a result of injecting the same amount of water into the test container simulating the canister of spent nuclear fuel and drying the test when the inner center temperature of the container reached 50°C, 100°C, 200°C, 300°C, and 400°C respectively, the residual moisture concentration was measured to be 2,200 ppmv to 2,400 ppmv. The residual moisture concentration at vacuum pressure of 7 Torr may vary slightly depending on the internal temperature or the flow rate control at measurement, and when the internal center temperature is about 100°C, the moisture concentration measurement value was $2,360 \pm 0.36$ ppmv.

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Keywords: Moisture quantification, Residual moisture, Dry storage, Spent nuclear fuel

A Review of Hydrogen Content Analysis Procedures for Spent Fuel Cladding

Myungchul Park*, Sunggeun Kim, and Dosik Kim

Korea Atomic Energy Research Institute (KAERI)

*mcpark@kaeri.re.kr

In the reactor core, Zr cladding reacts with coolant water, leading to the partial absorption of hydrogen produced as a byproduct of Zr oxidation. Since this hydrogen affects the mechanical properties of the cladding and its quantity shows the degradation level of the cladding, the analysis of hydrogen content in the cladding is classified as one of the fundamental post-irradiation examinations for spent fuel characteristics. The Post-Irradiation Examination Facility (PIEF) of the Korea Atomic Energy Research Institute (KAERI) is equipped with the equipment for analyzing the hydrogen content of spent fuel cladding, and this hydrogen analyzer has been used for evaluating spent fuels. In this paper, we summarize the basic operating principles of such equipment and review the procedures for handling and testing spent fuel cladding as radioactive materials. These improvements will be reflected in the procedure and the planning to upgrade the equipment for better experimental results in future tests.

Keywords: Hot cell, Hydrogen content analysis, Spent fuel, Zirconium, Fuel cladding

Optimization of Oxide Reducer Design for Spent Nuclear Fuel Pyroprocessing

Jinmok Hur* and Jae Soo Ryu

Korea Atomic Energy Research Institute (KAERI)

*jmhur@kaeri.re.kr

This study analyzes the current density distribution and heat transfer of the oxide reducer used in spent nuclear fuel pyroprocessing through COMSOL modeling. The substitution of conventional platinum anodes with carbon anodes in the oxide reduction process facilitates the application of high currents. However, it has been observed that the molten salt temperature in the reactor increases rapidly under high currents, necessitating the optimization of current density distribution of electrodes to maintain an optimal operational temperature range.

This research investigates the impact of conventional impermeable anode shrouds, typically used in laboratory-scale electrolytic reducers, and the effects of parallel multiple electrode configurations aimed at scaling up the electrolytic reducer. The findings highlight the need for a permeable anode shroud and parallel multiple electrode configuration to optimize the oxide reducer.

The study identifies salt-resistant heat and decay heat of fission products as significant contributors to heat generation during the operation of the oxide reducer. Under various heat generation conditions, the effects of electrode distance, argon gas flow rates, external convective heat transfer, and the incorporation of graphite plates for enhancing heat transfer in the oxide reducer were examined. The results indicate that, within the design limitations of the oxide reducer studied in this study, the necessity for salt purification can be delayed by at least threefold through the use of graphite plates for heat transfer.

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Keywords: Oxide reducer, Current density distribution, Heat transfer, Spent nuclear fuel, Pyroprocessing

Spent Nuclear Fuel Chlorination for the Separation of Strontium and Barium

Jinmok Hur* and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*jmhur@kaeri.re.kr

The separation of high heat load fission products, such as alkaline earth metals, from nuclear spent fuel can mitigate the challenges associated with spent fuel disposal. This study evaluates the feasibility of isolating strontium and barium from light water reactor spent fuel using non-aqueous processes. The proposed methodology involves heating the spent nuclear fuel to high temperatures to remove volatile nuclides, followed by chlorination with a chlorinating agent. The resulting chlorinated products are then treated with a precipitating agent in LiCl-KCl molten salt to achieve further separation. Subsequently, the remaining liquid undergoes distillation to recover strontium and barium. Thermodynamic equilibrium calculations were conducted to analyze the process flows.

Under the specified process conditions, both MgCl_2 and NH_4Cl chlorinating agents effectively converted SrO and BaO to SrCl_2 and BaCl_2 , respectively. Moreover, the precipitating agent Li_2CO_3 demonstrated superior separation efficacy compared to Li_3PO_4 . Thermodynamic calculations indicate that the strontium and barium recovered through MgCl_2 chlorination, Li_2CO_3 precipitation, and distillation contain 0.18%, 1.06%, and 0.32% impurities in terms of mass, radioactivity, and decay heat, respectively.

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Keywords: Chlorination, Thermodynamic equilibrium, Spent nuclear fuel, Separation, Alkaline earth metals

Preliminary Study on the Final Management of the Pretreated High-Level Wastes Based on the Mined Deep Borehole Matrix Concept

Heui-Joo Choi*, Jong Youl Lee, Jungho Hur, Hong Jang, and Hunsuk Im

Korea Atomic Energy Research Institute (KAERI)

*HJCHOI@kaeri.re.kr

Considering our country's limited natural resources, it is expected that we will have to rely on the nuclear energy for a significant portion of our energy for quite a long period of time. However, if we do not find a final management plan, the huge accumulation of spent nuclear fuel (SNF) due to the continued operation of nuclear power plants is inevitable. Currently, some countries are developing a direct disposal, and others the recycling technologies for the final management of SNF. So far, KAERI developed the relevant technologies in two directions, i.e. "direct disposal" based on the KBS-3 concept, and the other "recycling" based on the pyro-processing of SNF.

Recently, the authors reviewed alternative concepts for the direct disposal of SNF instead of KBS-3 concept for Korea. Among them, the concept of a mined deep borehole matrix disposal (MDBM) seems to be promising for the disposal of high-level waste (HLW) produced from the pretreatment of SNF since this method is suitable for the solidified waste with a relatively small diameter. According to the report from the UK Nuclear Decommissioning Authority [2008], AECL (Canada) and NUMO (Japan) both considered disposal of waste packages using MDBM as a potential option for hard rock. The concept can result in a relatively small repository footprint, due to the vertical extent of the repository, and a small excavated volume per waste package. Compared to deep borehole disposal (DBD), the technology for construction of MDBM is already standard.

In this paper, we conceptually design a MDBM disposal system for the HLW from the pretreatment of PWR SNF. Some sets of around 300 m long deposition boreholes with a diameter of 1 meter are drilled between the construction tunnel located at 800 m deep and the operation tunnel at 500 m deep for the disposal of the HLW. The dimension of a wasteform produced from pyro-processing is 30 cm in diameter and 30 m in height. A copper-cast iron disposal canister that can accommodate five wasteforms is designed. The copper canister is surrounded with compacted Ca-bentonite blocks. The MDBM concept is compared with the conventional deep geological disposal (DGD) of spent nuclear fuel and of HLW from a footprint perspective. The results show that the MDBM is quite competitive for the disposal of the HLW.

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Keywords: Spent nuclear fuel, Mined deep borehole matrix, Pretreatment for disposal, HLW, Disposal canister, Footprint

Manufacturing and Preliminary Evaluation of a Multi-array Electrolytic Recovery Electrode Module Using Automation Concept

Eunsoo Lee*, Sangkwon Lee, Chang Hwa Lee, and Jae Soo Ryu

Korea Atomic Energy Research Institute (KAERI)

*eslee@kaeri.re.kr

Designing an electrode module with automation concepts is necessary to facilitate the separation/connection processes for ease of operation. The electrolytic recovery device installed in PRIDE facility contained separated electrorefining, electrowinning, and residual actinide recovery devices, and the size of the devices required for the process was very large due to the arrangement of the reactor and electrode in the circular arrangement. As a way to solve this problem, a system was manufactured that integrates the electrolytic recovery process (electrorefining, electrowinning) so that it can be carried out in one device and reduces the overall size by introducing a multi-array module type electrode. In this study, moving beyond manual operation, we propose multi-array electrolytic recovery electrode module using automation concept through the design and fabrication of an electrode module. It is intended to improve process integration and operational efficiency.

A way to improve the efficiency of the existing electrolytic recovery process was established, and based on this, an electrolytic recovery electrode module using automation concepts was designed and manufactured. For the linkage between processes, the electrorefining anode and electroreduction cathode were designed with the same structure, and by designing the carbon electrode of the electrowinning and electroreduction process identically, the process device can be simplified. In order to enhance operation efficiency, the additional connection parts were installed on the top of the electrode modules so that it could be transported by the gantry automated handling device, and the upper and lower parts were designed to be separated by simplifying the bonding structure without bolts. Then the separated lower part allows the whole, including the crucible, to be loaded in the distillation process which is a subsequent process.

Prior to the preliminary evaluation of the manufactured electrode module, it is necessary to determine the feasibility of the existing multi-array electrolytic recovery device so the state of the salt in the device and feasibility of the system were confirmed. The salt was found to have a small amount of UO_2 impurities, but the amount was insignificant, and the concentration of UCl_3 was 2.3wt%, and as a result of conducting a U recovery experiment on a laboratory scale, it was judged that there would be no major problem. In addition, it is confirmed that there was no issues in the operation of the existing systems such as power supplies and scrapers in three experiments using existing electrode.

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Keywords: Pyroprocess, Electrolytic recovery process, Multi-array electrolytic recovery electrode module, Automation

Mechanical Integrity Analyses of High Burnup SNF Rod (PLUS7) Under Transverse Tearing Modes

Sang-Youn Jeon*, Joon-Kyoo Park, Jong-Sung Yoo, Seong-Ki Lee, and Nam-Gyu Park

KEPCO Nuclear Fuel (KEPCO NF)

*syjeon@knfc.co.kr

During the transportation of Spent Nuclear Fuel (SNF), there can be several events that could potentially affect the integrity of SNF. The normal condition and hypothetical accident conditions that may occur during transportation are defined to encompass those events. SNF has complex material properties resulting from in-reactor high temperature operation and exposure to radiation, making its mechanical behavior challenging to handle. Since 1990s, U.S. Department of Energy (DOE) and Sandia National Lab (SNL) have developed the methods for analyzing the mechanical response and integrity of SNF. They have identified failure modes of SNF during transportation as transverse tearing (Mode 1) and rod breakage (Mode 2) in the transverse direction, and longitudinal tearing (Mode 3) in the longitudinal direction, and found that radial hydride is relevant only in the longitudinal direction. The Electric Power Research Institute (EPRI) developed a method for analyzing mechanical behavior of high-burnup cladding using three-phase damage model, and determined the failure probability for each mode.

In this study, finite element analyses (FEA) of PLUS7 high burnup spent nuclear fuel rod under transverse tearing mode were performed for the mechanical integrity evaluation using ABAQUS/CAE 2018. The purpose of FEA is to calculate the bending moments and axial tensile forces for PLUS7 spent nuclear fuel rod. The dynamic forces affecting Mode-I/Mode-II failure sequences are the bending moments and axial tensile forces. The fuel rod finite element model was developed as a half model for the cross section of fuel cladding and pellet, and symmetry conditions were applied. There is a small gap between fuel cladding inner surface and fuel pellet outer surface to simulate the contact of fuel cladding and fuel pellet. The fuel rod finite element model consists of 4,299 nodes and 1,361 elements. Several axial displacements and rotations were applied as a load conditions. Based on the analysis results, several combinations of axial tensile forces and bending moments which cause the Mode-I/Mode-II failure of fuel rod were generated. The axial tensile forces and bending moments will be used to generate the generalized Strain Energy Density (SED) distribution. The SED distribution for the drop accident analysis results of transportation cask will be calculated and compared with the Critical Strain Energy Density (CSED) distribution to calculate the fuel failure probability for the reliability evaluation of spent nuclear fuel.

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Keywords: Spent nuclear fuel, Mechanical integrity, Finite Element Analyses (FEA), Zirconium hydride, Strain Energy Density (SED), Critical Strain Energy Density (CSED), Failure probability, Reliability evaluation

Evaluation of the Mechanical Properties of Stabilized Pellets From Damaged Nuclear Fuel

Yunmock Jung*, Kweonho Kang, Chang Hwa Lee, and Jae Soo Ryu

Korea Atomic Energy Research Institute (KAERI)

*jym@kaeri.re.kr

Damaged nuclear fuel refers to fuel that has undergone physical or chemical damage within a reactor. Stabilized pellets are the fuel generated in the process of processing and stabilizing spent nuclear fuel that has been recovered after removing its cladding. To safely manage and store these fuel, it is essential to accurately understand and evaluate their various mechanical properties. This study examined the differences in density based on burnup and compaction pressure of stabilized pellets, as well as the differences in mechanical strength through measurements of compressive strength and elastic modulus.

Specifically, stabilized pellets were produced with burnup of 0, 35, and 55 GWd/tU, compaction pressures of 300 MPa and 350 MPa, and sintering temperatures of 1,550°C and 1,700°C. The mechanical properties, such as density, compressive strength, and elastic modulus, were then measured and analyzed. For sim fuel (Simulated nuclear Fuel), the density increased with higher sintering temperatures, while it decreased with higher burnup. Compressive strength and elastic modulus increased with increasing density but decreased with increasing burnup. As the sintering temperature increased from 1,550°C to 1,700°C, the compressive strength of UO₂ increased from 717.8 MPa to 897.4 MPa. For 35 GWd/tU, it increased from 427.1 MPa to 732.6 MPa, and for 55 GWd/tU, it increased from 301.4 MPa to 515.5 MPa. Additionally, for simulated nuclear fuel sintered at 1,700°C, as the burnup increased from 0 to 55 GWd/tU, the density decreased from 10.58 g/cm³ to 9.99 g/cm³. As the compaction pressure of the stabilization pellets increased from 300 MPa to 360 MPa, the relative density increased from 9.37 g/cm³ (87.81% of TD) to 9.45 g/cm³ (88.56% of TD), and the compressive strength increased from 569.5 MPa to 590.72 MPa. Additionally, the elastic modulus slightly increased from 9.14 GPa to 9.16 GPa.

The mechanical data of the stabilization pellets produced in this study can serve as reference data for the development of stabilization technologies for damaged nuclear fuel, as well as for the integrity assessment during transportation, storage, and disposal. Furthermore, these findings can be utilized as fundamental data to increase public acceptance of spent nuclear fuel management.

Keywords: Stabilized pellets, Damaged nuclear fuel, Compaction pressure, Compressive strength

The Effect of Hygroscopic Fission Products on the Density of High Burnup Nuclear Fuel Pellets

Yunmock Jung*, Kweonho Kang, Jaesun Han, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*jym@kaeri.re.kr

The density of nuclear fuel sintered pellets tends to decrease with higher burnup. A decrease in the density of nuclear fuel pellets can cause a variety of thermal, mechanical, and radiological safety issues in reactor operations. The reasons for the decrease in density include the accumulation of FPs (Fission Products), the release of fission gases, swelling, structural changes, and the formation of defects/cracks.

In addition, certain FPs are hygroscopic and can absorb moisture, affecting the physical and chemical properties of the pellets. Therefore we selected La (Lanthanum) which is one of the fission products that is hygroscopic, and the research aimed to determine how much the density changes with variations in La content. We used CeO_2 as a surrogate for UO_2 and La_2O_3 as a surrogate for La. The green body was molded using a single action press and the specimen weighed 2.5 ± 0.1 grams. As a comparison, samples were prepared using only CeO_2 powder at four different compaction pressures (70, 80, 90, 100 kgf/cm^2). The density and diameter change rate were measured according to the compaction pressure. Additionally, to investigate the density change with varying La content, CeO_2 and La_2O_3 were mixed in ratios of 99:1, 97:3, 95:5, and 90:10, and then compacted at a pressure of 90 kgf/cm^2 . The green body were exposed to air for 24 hours and then sintered at $1,600^\circ\text{C}$ for 5 hours.

Finally, the pellets density and diameter change rate were measured by geometrical method. As a result, when pellets were made using only CeO_2 , the density increased proportionally with the compaction pressure, reaching 89-91% of TD (Theoretical Density). It was also observed that the diameter shrank by approximately 12-13%. However, the density of sintered mixtures of CeO_2 and La_2O_3 ranged from 40% to 74% of TD. The diameter showed a shrinkage of 3-7% for the 99:1 and 97:3 mixing ratios, while for the 95:5 and 90:10 mixing ratios, it actually expanded by approximately 4-15%.

In general, hygroscopic FPs within 1% are generated in high burnup nuclear fuel. The observed differences of approximately 10% in TD and about 5% in diameter indicate that the hygroscopic of FPs can also be a significant factor for assessing the integrity of high burnup nuclear fuels.

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Keywords: Hygroscopic, Fission products, Lanthanum, High burnup pellets

A Study on the Stress State of Fuel Rod for the Bonding Ratio Between Pellet and Cladding Under Normal Conditions of Transportation

Jae-Jun Lee* and Seong-Ki Lee
KEPCO Nuclear Fuel (KEPCO NF)
*jaejunlee@knfc.co.kr

A lot of Spent Nuclear Fuels (SNFs) have been stored in SNF pools, and the pools may be saturated in the near future. To prevent saturations of the pools, the dry storages for the SNFs shall inevitably be necessary, and the transportations of SNFs are essential for the dry storages. The generated loads from external factors are transferred to fuel rods in SNF during the transportation. Also, the bonding condition is the main boundary condition affecting the fuel rod stiffness in SNF. In this study, the stress states of fuel rod are evaluated for bonding ratios between pellets and cladding under Normal Conditions of Transportation (NCT).

The fuel rod stresses for each bonding ratio were calculated from Finite Element (FE) analysis. The FE model was generated based on 3 spans of fuel rod, and the both ends of the model were fixed as boundary conditions. The loading conditions consist of external and internal load condition, respectively. The external loads applied the actual load spectrums, which are the NCT test results using surrogate fuel based on SNF conditions. Internal load applied the internal pressure in fuel rod of SNF, and the pressure was calculated using the stress equation of cylindrical pressure vessel and the hoop stress (90 MPa) presented in the regulatory conditions.

The reference temperature of the cladding in FE analysis was set to 400°C, because that temperature is the highest temperature among the series of processes for dry storage and occurs during the drying process. The 400°C cladding properties were produced by linear interpolation based on tensile test results (Room temperature, 300°C) using the irradiated claddings conducted by EPRI (U.S. Electric Power Research Institute).

Inner surface of the cladding and outer surface of pellet are bonded due to degradation during burn-up and NCT, etc. However, it is difficult to know the bonding state between the two components. Thus, arbitrary bonding states were generated in the FE model. The bonding ratios were set from 0% (non-bond) to 100% (perfect bond) in 10% increments, and the bonded nodes of pellets were selected by generating random numbers. Two FE analyses for each bonding ratio were performed based on the ascending and descending order of the random numbers.

From the analyses, the stress level was similarly calculated from 0% to 80% of the bonding ratio. After exceeding 80%, the stress for each ratio decreased rapidly due to the interaction of bonding. The maximum stress was calculated as 1.09S at 30% ratio, and the minimum stress was calculated as 0.1S at 100% ratio: the reference for non-dimensional stress is the stress (1.0S) at a ratio of 0%. Therefore, the fuel rod evaluation under NCT conditions can be conservatively evaluated by using 110% value of the stress for the non-bonded condition between pellets and cladding.

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Keywords: Spent Nuclear Fuel (SNF), Bonding ratio, Pellet, Cladding, Normal Conditions of Transportation (NCT), Finite Element (FE) analysis

Strategy to Development of Concrete Dynamic Material Model for Dry Cask Storage System

JaeHoon Lim*, Jongmin Lim, Yun-young Yang, Seung-Pil Kim, and Sang Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*jhl85@kaeri.re.kr

Concrete structures of spent nuclear fuel interim storage facility should maintain their ability to shield and structural integrity during normal, off-normal and accident conditions. The concrete structures may deteriorate if the interim storage facility operates for more than several decades. Even if deterioration occurs, the concrete structures must maintain their own functions such as radiation shielding protection and structural integrity. Therefore, it is necessary to establish an analysis methodology that can evaluate whether the deteriorated concrete structure maintains its integrity under not only normal or off-normal condition but also accident condition.

In this study, a strategy for the development of a concrete dynamic material model for dry storage facilities was established. First, the well-known Popovics concrete material model was analyzed. The results were then compared with the dynamic material testing results of HANARO concrete. The results showed that the Popovics model, which is a model for the static properties of concrete, differed from the dynamic test results. The dynamic material testing of HANARO concrete revealed that as the strain rate increased, not only did the maximum stress increase, but the strain value at which maximum stress occurred also increased. Therefore, a dynamic material model for concrete should be able to predict not only changes in maximum compressive stress according to strain rate but also changes in strain at maximum compressive stress due to strain rate. The dynamic material model will be developed in conjunction with degradation parameters (e.g., chloride penetration, freeze-thaw cycles). Once the dynamic material model is completed, it will provide stress-strain diagrams according to the strain rate based on the static material properties.

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Keywords: Dry cask storage system, Interim storage facility, Dynamic material testing, Degradation

Evaluation of Radiation Dose Rates by Location for Worker Transporting Spent Nuclear Fuel

Shin Dong Lee, Dong Gyu Lee, Jeong Woo Lee, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

In Korea, Spent nuclear fuel (SNF) will be managed through the stages of temporary and interim storages before final disposal. In this process, workers at the storage facility must perform the transport of SNF from the current stages to the subsequent stages, and radiological workers who transporting SNF are subject to ALARA principles.

Radiation dose rates to workers handling SNF loaded in casks depend on the location of work. Radiation dose assessment for workers transporting SNF can be performed by considering the dose rates by location of work, number of workers, duration time to work. Therefore, this study evaluated radiation dose rates by location of work for transporting SNF.

To evaluate the radiation dose rates by location of work for transporting SNF, assumptions were established first. Then, the locations of work for transporting SNF were established. Finally, radiation dose rates by location of work for transporting SNF were evaluated. Assumptions were established for casks to transport SNF, vehicle for transfer and transportation etc. For location of work, operational procedures specified in SAR of HI-STAR 100, HI-STORM 100 commercially used in foreign countries were used as reference. Since the specific distances for each location of work were not specified in the SAR, the distances for location of work were assumed, reflecting the tendency of work. Finally, the MCNP computer code was utilized to simulate the cask and location of work, and radiation dose rates by location of work were derived.

As a result of this study, a total 5 locations of work were established for workers transporting SNF. 3, 2 locations were set for the situation with and without impact limiter, respectively. Most of the location of work were set at 10 cm from the surface of the cask. Transfer cask through vertical cask transfer, loading cask on the transportation vehicle, performing a visual inspection of cask were set at 100 cm from the surface of the cask only. The evaluation of the radiation dose rates at each location showed that the highest dose rate was evaluated at the bottom of the cask before impact limiter was installed, and lowest radiation dose rate was evaluated at the surface of the impact limiter.

This study evaluated the radiation dose rates by each location of work for transporting SNF. The results of this study are expected to be used in the future to assessment of the radiation dose to workers transporting SNF.

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Keywords: Spent nuclear fuel, Transporting, Location of work, Radiation dose rate, Operational process

Design of Test Equipment for Thermal Analysis Evaluation of Dry Storage Demonstration System for Spent Fuel in CANDU Reactors

Changhwan Shin*, Dae-ho Kim, and Dong-hak Kook

Korea Atomic Energy Research Institute (KAERI)

*shinch@kaeri.re.kr

Spent nuclear fuel from CANDU reactors is stored in dry storage facilities, such as canisters or MACSTOR, before permanent disposal. The canister-type dry storage facility has been in operation since 1992, totaling over 30 years. There is an increasing need for a demonstration system to assess the integrity of this nuclear fuel and to secure evaluation technologies. The primary factor affecting the degradation mechanisms of the fuel in dry storage is its temperature. Although decay heat significantly decreases after several years of wet storage, heat generation continues, making thermal analysis crucial in sealed containers. The concrete canister consists of nine metal baskets containing 60 fuel assemblies and relies on natural convection for cooling.

Temperature analysis of the canister is often conducted using computational fluid dynamics (CFD) techniques, which will also be applied in the thermal analysis of the demonstration system. This paper describes the test equipment designed for the evaluation of thermal analysis. The test setup includes a Single Fuel Assembly Test (SEPA_F) for handling a single assembly, a Single Basket Experiment (SEPA_B) for evaluating the temperature field within a basket. And an Insulation Condition Experiment (ASSEM_A) for the stacked baskets, and a Canister Demonstration Test (ASSEM_C) simulating the concrete structure.

The Single Assembly Test aims to assess heat transfer characteristics and the performance of a cylindrical heater simulating the fuel assembly in the basket tests. The Single Basket Experiment focuses on evaluating the temperature field based on the power distribution of the fuel assembly. The Insulation Condition Experiment will assess the flow and temperature fields generated by natural convection between the liner and the basket while insulating the liner's outer wall. The Canister Demonstration Test will evaluate the temperature characteristics inside the basket influenced by external temperatures. The data accumulated from these experiments will validate computational models for assessing the temperature history of nuclear fuel in dry storage systems, thereby enhancing model reliability.

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Keywords: Dry storage, Demonstration system, CANDU, Canister, Basket, Temperature field

Holistic Approach of Similarity Analysis Between Canadian and Korean CANDU Spent Fuel Characteristics in Dry Storage Condition

Donghak Kook*, Yongsik Yang, and Daeho Kim

Korea Atomic Energy Research Institute (KAERI)

*syskook@kaeri.re.kr

The first Korean CANDU spent fuel dry storage system called as ‘Silo’ has been operated from 1992 under 50 years operation license. Regardless of successful siting of spent fuel management like central dry storage (away-from-reactor) or repository by the government, CANDU spent fuel characteristic analysis is necessary for following both cases. First, if dry storage operation is extended at the original place beyond initial license duration, storage system operator has to check and show spent fuel’s integrity for extended storage duration. Second, if spent fuel is transported to away-from-reactor site like central dry storage or repository, spent fuel management implementor from the government has to check the spent fuel’s condition before long distance dynamic transportation.

Most spent fuel characteristic analysis rely on hotcell testing data, but unfortunately Korea has no experience on CANDU spent fuel hotcell testing. Therefore, a holistic approach to build up CANDU spent fuel data by several steps is undergoing in spent fuel dry storage demonstration R&D program. First, ductility test with fully oxidized unirradiated cladding in dry condition is to understand mechanical behavior analysis. Second, in-situ measuring spent fuel by non-destructive examination under reactor spent fuel pool is to understand spent fuel characteristics before drying and dry storage operation. Third, acquisition of Canadian hotcell data for dry storage is to understand basic and general specifics of irradiated material testing results. Fourth, comparison between Korean and Canadian spent fuel characteristics including dimensions and other features is to analyze similarity and apply it to Korean CANDU spent fuel characteristic analysis system.

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Keywords: CANDU spent fuel, Dry storage, Spent fuel characteristics, Hotcell, Similarity, Long-term integrity

Separation of High Heat Generating Nuclide in Agitated LiCl-CaCl₂ Molten Salts at 650°C

Sang-Kwon Lee* and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*sklee79@kaeri.re.kr

This study investigates the separation efficiency of high heat generating nuclides in LiCl-CaCl₂ molten salt systems, with a particular focus on the effect of agitation to enhance the dissolution reaction. Previous research has identified MgCl₂ as a promising molten salt for the separation of Sr from simulated oxide fuel at 800°C. In the present work, we explore the dissolution behavior of nuclides in LiCl-CaCl₂ molten salt at 650°C, aiming to reduce the operating temperature while maintaining high separation efficiency. The composition of the simulated oxide fuel comprised 2wt% Sr, 3wt% Ba, 2wt% Ce, 3wt% Nd, 3wt% Zr, 2wt% Mo, and 89wt% U. To measure the Sr concentration in the salt, ICP analysis was employed following salt sampling using a dip-stick technique. Additionally, ICP-OES analysis was conducted to evaluate the separated nuclides. Complementary to these techniques, SEM and XRD analyses were performed to confirm the crystal structure and morphology of the oxide products.

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Keywords: Agitation effects, LiCl-CaCl₂ molten salt, High heat generating nuclide separation, Simulated oxide fuel

Analysis of PIE Data for CANDU Spent Fuel in the Dry Storage Environment

Dae-Ho Kim^{1,*}, Dong-Hak Kook¹, and Tae-Hyung Na²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kdh@kaeri.re.kr

The spent fuel from the domestically operated CANDU nuclear power plants is stored in interim dry storage facilities, such as silo and MACSTOR. The dry storage facility for CANDU spent fuel, known as the concrete silo, has been operated since 1992 and was licensed with a design life of 50 years. To achieve this, it is necessary to evaluate the integrity of both the silo and the spent fuel to demonstrate safety through a series of assessments. However, existing data on the integrity of the current silo and the performance of the CANDU spent fuel is difficult to find. By reviewing the post-irradiation examination (PIE) data of CANDU spent fuel obtained from Canada, we aim to predict the status of CANDU spent nuclear fuel in a domestic dry storage environment.

An analysis was conducted on the PIE data of spent fuel used in a dry storage demonstration test in Canada. The dry storage demonstration test involved placing actual spent fuel in a test container, then periodically removing it for PIE at various intervals. The PIE designated standard fuel rods from representative fuel bundles, and destructive & non-destructive tests were conducted after specific periods (41, 100, 140, and 200 months) by removing adjacent spent fuel rods.

The PIE data, including the appearance of the spent fuel cladding, mechanical properties of the HAZ/ARZ, endcap/endplate torque, hydrogen and deuterium (H/D) concentrations, and UO₂ oxidation behavior, were analyzed according to the storage period in a dry storage environment. The comprehensive analysis concluded that the CANDU spent fuel maintained its integrity well in the dry storage environment.

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Keywords: PHWR, CANDU spent fuel, Post-irradiation Examination (PIE), Dry storage, Concrete silo, MACSTOR

Temperature-Dependent Ring Tensile Properties of Cr-Coated Zircaloy-4 Cladding for Accident Tolerant Fuel (ATF)

Dae-Ho Kim*, Hong-Ryoul Oh, Jong-Dae Hong, and Hyun-Gil Kim

Korea Atomic Energy Research Institute (KAERI)

*kdh@kaeri.re.kr

Accident Tolerant Fuel (ATF) cladding is being developed by applying various technologies to ensure safety, economic efficiency and reliability. The coating concept on zirconium-based alloy tube with various oxidation-resistance materials is being proposed and developed to reduce hydrogen generation at high temperatures, and maintain mechanical integrity of the cladding, thereby increasing the safety margin during accidents.

Currently, ATF cladding concepts are being streamlined to Cr and CrAl-coated claddings, with Arc Ion Plating (AIP) coating methods. The characteristics of the coating layer vary depending on the coating target and process, and deformation may occur due to the electrical and thermal effects exerted on the base material by the coating material. Therefore, it is necessary to establish optimal coating process conditions that ensure the mechanical integrity of the coated cladding and to verify performance through mechanical property testing.

Ring tensile tests of Cr-coated Zircaloy-4 cladding, coated at the lab scale, were conducted at room temperature, 300°C, 360°C, 400°C, 600°C, 700°C and 800°C. The thickness of the chromium coating layer on the cladding was 15 μm . The ring tensile strength at room temperature was 831 MPa, slightly higher than the 813 MPa of the as-received cladding, and the change in tensile strength with increasing temperature was similar to that of the as-received cladding. No delamination or abnormal separation of the coating layer due to artificial loading was observed. Microstructural observations through OM indicated that the laminated state of the coating layer was satisfactory. The base material retained an SRA structure, suggesting that there was no thermal impact from the coating process. It is judged that the optimal coating process has been established from the perspective of manufacturing feasibility of the Cr-coated cladding through this ring tensile test, and that the appropriate tensile strength has been shown at each temperature.

Acknowledgements

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Keywords: Accident Tolerant Fuel (ATF), Arc Ion Plating (AIP), Cr-coated cladding, Ring tensile test, Cr coating layer, SRA

Integrity Assessment of Heavy Water Reactor Spent Fuel Bay Under Seawater Ingress

Donghee Lee

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

donghee.lee@khnp.co.kr

In June of this year, during scheduled preventive maintenance at a heavy water nuclear power plant, a coolant leak occurred in the spent fuel storage bay, resulting in the release of coolant into the ocean. Based on the environmental impact assessment conducted on the leaked quantity, the effective dose was found to be significantly below the annual dose limit for the general public. Additionally, seawater samples were taken from the internal drainage system and nearby marine areas to check for specific radionuclides, none of which were detected.

To prevent the recurrence of a similar incident, an evaluation was conducted assuming a scenario in which seawater might ingress into the spent fuel storage bay if the system pressure is modified such that coolant leakage is prevented. The results of this assessment indicated that, under seawater conditions, the radiation dose rate within the heavy water reactor's spent fuel storage bay would decrease by approximately 14% compared to the original coolant (water) environment.

Keywords: Heavy water reactor, Spent Fuel Bay (SFB), Seawater

Radiation Dose Assessment by Neutron Shield Material Configuration in Spent Fuel Transfer Casks

Donghee Lee*, Taehyung Na, and Taehyeon Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*donghee.lee@khnp.co.kr

The spent fuel (SF) transfer and storage casks are designed with a basket structure that supports the SF and prevents criticality by absorbing thermal neutrons using neutron absorbers like B₄C. These casks consist of a containment section that separates the SF storage area from the external air layer and an external shielding layer to block radiation. The external shielding layer is typically composed of high-density metals such as iron or lead for gamma radiation shielding and materials containing B₄C or concrete for neutron shielding.

Recently, there has been a trend towards transporting or storing SF at an earlier stage, soon after its generation. The challenge with this approach is that the increased decay heat requires higher thermal removal capacity from the transportation and storage casks. Traditional neutron shielding materials, such as resin-based substances like NS4FR or concrete, generally have low thermal conductivity. Resin-based materials, in particular, have a melting point below 150°C, which poses risks in the event of a fire and contributes to a reduction in the thermal capacity limit of the cask design due to their poor thermal performance. To address this issue, there is a need to develop neutron shielding materials that can maintain performance during a fire and enhance the thermal capacity of storage casks.

Therefore, this paper aims to develop a neutron shielding material by proposing a novel neutron shield that utilizes graphite, which moderates fast neutrons by converting them into thermal neutrons, and boral, which is highly effective at absorbing thermal neutrons. The study investigates the shielding performance based on the arrangement of these two materials.

Keywords: Spent fuel transfer casks, Neutron shield material, Radiation dose assessment

Review of the Analysis Report on the Inventory of Spent Nuclear Fuel and Reprocessing Waste in the United States

Donghee Lee* and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*donghee.lee@khnp.co.kr

Pacific Northwest National Laboratory (PNNL) has released a new report titled, “Spent Nuclear Fuel and Reprocessing Waste Inventory”. This report details the inventory of spent nuclear fuel (SNF) located at nuclear power reactors and independent spent fuel storage installations (ISFSIs) in the United States, as well as spent fuel and reprocessing waste located at US Department of Energy (DOE) sites and other research and development (R&D) centers as of the end of calendar year 2022. The report also includes inventory forecasts for spent fuel for a few selected scenarios of future nuclear power generation from the existing reactor fleet, and reactors under construction for one case. It also includes select information on the characteristics associated with the wastes examined, such as the type, packaging, heat generate rate, decay curves.

The source of historical inventory is the Nuclear Fuel Data Survey collected by the Department of Energy on the GC-859 forms.

As of the end of 2022, the US inventory of spent nuclear fuel and primary reprocessing waste was located at over 100 sites in 39 states, including. To develop the inventory, estimate beyond 2022, spent fuel discharge projects were developed using the US Commercial Spent Nuclear Fuel Projection tool [Vinson, 2015]. The estimate of future spent fuel inventory assumes 60 or 80 years of operation for existing reactors. The reference scenario assumes no new nuclear power reactors are built or will shut down in the future.

By analyzing the technologies used in the United States to manage the inventory of spent nuclear fuel and reprocessing waste, we aim to apply these insights to the future evaluation of spent nuclear fuel and waste inventory in Korea.

Keywords: Spent fuel transfer casks, Neutron shield material, Radiation dose assessment

Cold Spray Metallic Coatings on Stainless Steel 304 for Mitigating Chlorine-Induced Stress Corrosion Cracking (CISCC) in Spent Nuclear Fuel Interim Storage

Jinwook Choi and Hwasung Yeom*

Pohang University of Science and Technology (POSTECH)

*hyeom@postech.ac.kr

Due to its urgency of deployment of Dry Cask Storage System (DCSS) in the Republic of Korea, stainless steel 304 L is planned to be employed rather than stainless steel 316 L as a canister material. However, stainless steel 304 L is more susceptible to pitting corrosion compared to stainless steel 316 L. During the manufacturing of DCSS, a Heat Affected Zone (HAZ) may occur when the ends of the rolled stainless steel plates are welded together. The HAZ is an area where chromium is depleted due to the precipitation of chromium carbide at the grain boundaries. Therefore, the depletion of chromium leads to a deterioration in corrosion resistance. This phenomenon is called material sensitization. When material sensitization, a chloride-rich (i.e., corrosive) coastal environment, and residual tensile stresses from the solidification and contraction of weld metal combine, pitting corrosion can occur and may develop into Chloride-Induced Stress Corrosion Cracking (CISCC). Cold spray coating technology is gaining attention as a treatment that can mitigate both pitting corrosion and CISCC. In this study, candidate materials such as stainless steel 304 L, 316 L, Ti, and Ni powders are coated on stainless steel 304 specimens using the cold spray coating technique. Electrochemical corrosion tests such as potentiodynamic anodic polarization tests, immersion tests, microstructural characterizations, and X-ray diffraction analysis of powders and coated specimens are being conducted to evaluate the composition of the raw materials, coating adhesion, and corrosion resistance of the coated samples.

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Keywords: Cold spray coating, Spent nuclear fuel, Dry cask storage system, Chlorine-induced stress corrosion cracking, Stainless steel 304

Spent-Fuel Transportation Applications of Normal Conditions (EPRI)

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

A comprehensive methodology was developed to calculate the global dynamic forces acting on the fuel rods and guide tubes in a spent fuel transportation cask with no impact limiters subjected to a 0.3-meter drop onto a rigid surface in the horizontal orientation. The modeling approach adopted for the high-burnup spent fuel is an extension of the approach used in the well-known source-term study by Sandia National Laboratory in the late eighties and early nineties, in which fuel rod failure probabilities were calculated for spent fuel in the low to intermediate burnup range subjected to normal and hypothetical accident conditions of transport. The horizontal drop condition in the Sandia study produced higher forces than either of the end drop or the corner drop orientations, which motivated the choice of the horizontal orientation for the present study.

The finite element modeling details emphasized fuel rods and guide tubes responses at locations where maximum forces would be expected, which include spacer grids at assembly mid-span and end-span positions. In order to capture the whole range of variation of dynamic forces in the cask, several detailed models of the assemblies were positioned at various locations in the basket structure: at the bottom where the forces would be largest, at the top where the forces would be smallest and at the side position where intermediate level forces would be expected. The dynamic forces of significance to fuel assembly damage evaluation consist of pinch forces caused by rod-to-rod impact, bending moments, and axial-extension forces.

Temporal and spatial variations of the forces acting on the fuel rods were calculated and post-processed to obtain frequency distributions, which statistically represent the total rod population in the cask. The largest pinch force is 1,400 lb, but over 90% of the rods experience pinch forces less than half this maximum. The maximum forces for the fuel rods are: 1,400 lb pinch force, 600 lb axial force and 175 in-lb moment. The maximum forces calculated for the guide tubes, which are the structural elements of the fuel assembly, are: 640 lb pinch force, 415 lb axial force and 348 in-lb moment. These forces are used in a detailed failure analysis procedure, the results of which indicate adequate margin against violating the regulatory limits for normal conditions of transport.

Keywords: Spent-fuel, Transportation, Normal conditions, Cask, Pinch force

NRC Approval to Transport HALEU TRISO Fuel in OPTIMUS-L

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

NAC International announced January 9 it has received approval from the Nuclear Regulatory Commission for its OP-TIMUS-L packaging system to store and transport high- as-say low-enriched uranium (HALEU) TRISO (TRi-structural ISOtropic particle) fuel, which will be used in advanced reactors. NAC noted that this is the first time the NRC has licensed high-capacity packaging greater than 500 pound payload – for HALEU TRISO fuel in the US. This certification lays the foundation for future licensing of NAC’s OPTI-MUS systems to carry other HALEU contents. The NRC issued the Certificate of Compliance (CoC) No. 9390, Rev. 2, on December 14, 2023. NAC can now transport HALEU TRISO fuel compacts to support an advanced reactor project for a customer.

Each TRISO particle consists of a uranium, carbon, and oxygen fuel kernel encapsulated by three layers of carbon- and ceramic based materials that prevent the release of radio-active fission products. HALEU TRISO fuels are considered to be the preferred fuel in several advanced reactor designs currently under development.

For over two years, Energy Northwest has investigated numerous emerging nuclear energy technologies and has concluded that X-energy’s advanced small modular reactor, the Xe-100, is the design best tailored to the region’s specific needs. “The ongoing analysis of project structures, schedules, and overall liability will inform future project investments and ultimately the prudence of a final investment decision”, Energy Northwest said in its January 10 press release.

Keywords: HALEU, Transportation, OPTIMUS-L, Cask

DOE's Plan for Producing HALEU Fuel From EBR-II

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

The US Department of Energy announced that spent nuclear fuel from the Experimental Breeder Reactor-II (EBR-II) will be recycled as part of a larger effort to make high-assay low-enriched uranium, commonly referred to as HALEU, more accessible to reactor developers to test and demonstrate their designs. EBR-II shut itself down from full power in 1986 without any operator action, in spite of a series of tests that were designed to see if the small sodium-cooled reactor would melt down. It did not, and the data used from the experiments are being used more than 30 years later to advance fuels and materials research for new reactor technologies. Now, DOE is using EBR-II to further develop and deploy advanced reactors by using the reactor's spent nuclear fuel. For over two years, Energy Northwest has investigated numerous emerging nuclear energy technologies and has concluded that X-energy's advanced small modular reactor, the Xe-100, is the design best tailored to the region's specific needs. The ongoing analysis of project structures, schedules, and overall liability will inform future project investments and ultimately the prudence of a final investment decision.

DOE is also supporting other efforts to provide more access to HALEU, including recycling spent fuel from other government-owned research reactors, supporting the demonstration of HALEU production at the American Centrifuge Plant in Piketon, Ohio, and acquiring HALEU through purchase agreements with domestic industry partners to spur demand for more HALEU production and provide investment in the US fuel supply infrastructure.

Keywords: HALEU, EBR-II, OPTIMUS-L, Sodium-cooled reactor, Spent fuel

CEX-1 Bundles Pre-Storage Post Irradiated Examination Results

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

During the pre-dry storage (baseline) examination, all of the outer elements were intentionally defected on the bundle, except control outer element #1. This strategy allowed for continual comparison to be made of storage effects on defected CANLUB-coated elements with intact CANLUB-coated elements. Intentionally-defected, CANLUB coated PNGS bundle 11171W arrived in the hot cell facility at Whiteshell Lab. on 1979 December 12 for baseline (i.e., pre-dry storage) examination, after 82 months of underwater cooling time.

Non-defected CANLUB-coated BNGS bundle F06584C was transferred to the Whiteshell hot cell facility on 1979 for the baseline (pre-storage) examination after 35 months of cooling time. Intentionally-defected CANLUB-coated BNGS bundle F066605C was also transferred to the Whiteshell hot cell facility on 1979 July 31 for the baseline (pre-storage) examination after 35 months of cooling time. Figure 4 is an end view of the bundle showing which (shaded) elements underwent baseline examination from bundle F066605C.

Fuel element “cutting diagrams” indicate the examination locations on selected fuel elements that were destructively examined from bundles 11171W, F06584C and F066605C during the Baseline examination.

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Keywords: Post irradiated examination, Pre-storage, Baseline, Bundle

Interim Storage Examination of CEX-1 Bundles (ISE-3)

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

The third (ISE-3) began when the CEX-1 basket was retrieved from storage on 1993 March 8 and unloaded on 1993 April 2 after approximately 140 months of storage (~100 months in limited air and ~40 months in “unlimited air”) at 150°C. Only fuel from the four defected bundles was examined during this ISE. Intact control element 1 and defected control element 2 were non-destructively examined from each bundle and returned to the CEX-1 basket and it was returned to storage on 1993 September 14. One outer element from each defected bundle was non-destructively examined in a similar manner, prior to a detailed destructive examination.

Analysis of the storage atmosphere of the compartment containing the four intact bundles indicated oxygen depletion to 18.3 mol% and ~0.16 mL of Xe gas in the compartment. Because of the small amount of fuel that could have been oxidized, no elements from intact bundles were examined. The condition of the intact control elements from the defected bundles appeared unchanged from the baseline examination.

In the defected elements, the amount of oxidation increased with storage time. At CEX-1-ISE2, up to 4% of the fuel in one element had oxidized to U₃O₇, however at CEX-1-ISE-3, up to 100% of the fuel had oxidized to U₄O₉. Traces of U₃O₈ were detected in three of the four elements. Evidence of UO₂ grain separation, intergranular cracking, and grain micro-cracking was observed with SEM, thought to be from an increase in fuel density from UO₂ to U₃O₇/U₄O₉. The profilometry showed no change in element diameter and gamma scans showed no significant change in Cs activity, although some migration was observed in some instances. The ultimate tensile strength of the sheath was lower than pre-storage values in the Pickering 28-element bundles, but the same or higher in the Bruce 37-element bundles.

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Keywords: ISE, CEX-1, Destructive examination, Bundle

The Dry Storage Silo (Canister) Inspection Procedures

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

Remote visual inspection (RVI) of the dry fuel storage canisters is to be developed for Silo (Canister). The inspections are carried out using the custom RVI tool developed by KHNP and Canadian Nuclear Laboratories (CNL). Inspections using the RVI tool are performed according to the Tool Operation and Maintenance Procedure.

Other services and service providers (e.g. crane, radiation protection, scaffolding, etc.) are involved in performing the dry storage inspections. This Work Plan serves as the principal document to specify the sequence of activities, hazards, procedures, and personal protection equipment & clothing (PPE&C) that are involved.

This Work Plan covers the activities required to perform the remote visual inspection of the dry fuel storage canisters using the RVI tool developed by KHNP & CNL. The inspections consist of visual inspection, temperature measurement, and radiation measurement. The RVI tool is deployed by crane onto the dry storage canister. A scaffolding system must be erected around the dry storage canister being inspected for personnel to anchor the RVI tool in position, and to install the various inspection tools on the RVI system. When an inspection tool is not in use it will be stored on its respective strongback, which is to be suspended from the scaffolding. The steps outlined in this Work Plan represent those for inspection of a single dry storage canister.

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Keywords: Silo, Storage canister, Basket, Remoter inspection system

Operating and Maintenance Procedures of the RVI

Taehyung Na*, Donghee Lee, and Kyungho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*taehyung.na@khnp.co.kr

This paper describes the assembly, operation, and disassembly of the remote visual inspection (RVI) tool that is used for inspecting the contents of the dry fuel storage canisters. The purpose of the RVI tool is to visually inspect the fuel baskets and the canister's liner while also measuring the temperature and radiation field within the canister.

The equipment to be deployed for the inspection of the canisters is comprised of Work platform (scaffolding) around the canister, Delivery system, Inspection tools and Control system. The developed Operating and Maintenance Procedures introduce the system and safety information for each part of the RVI and a comprehensive procedure for Equipment Assembly, Initial Commissioning, Installation and Operation was developed and described. In addition, the equipment can be disassembled and stored in a dedicated container at any time, and procedure is introduced. It describes how Silo Mock-up can be installed and utilized to prepare the operator for future inspection of the actual canister.

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Keywords: Silo, Remoter inspection system, Maintenance, Operating procedures

Structural Safety Assessment of Dry Storage System Canister

Taehyung Na

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

taehyung.na@khnp.co.kr

Canisters are used to extract spent fuel stored in wet storage tanks in nuclear power plants and transport and store it in dry storage facilities. Canisters loaded with spent nuclear fuel in wet storage tanks close the lid, install the sealing ring, and perform multiple sealing by the second welding.

After that, the moisture inside is removed and converted to the dry storage environment. The canisters are stored after being transferred to the dry storage facility using the transfer container or the transfer container. The soundness of the canisters under the transfer conditions is carried out in the evaluation of the transfer container, and the main purpose of this study is to design the canisters under the internal storage conditions of the dry storage facilities and to secure reliability through the structural integrity evaluation based on the code requirements.

The load application requirements of the structural analysis for the structural integrity assessment were based on the containment boundary and internal composition requirements of the canister to assess the soundness of the canister designed for normal, abnormal, and accident conditions.

Keywords: Canister, Silo, Spent fuel, Structural safety assessment, Dry storage system

Simultaneous Distillation of Rare Earth-TRU-Alkaline Earth Chlorides

Sang Woon Kwon*, Eunsoo Lee, Byungsuk Park, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*swkwon@kaeri.re.kr

Alternative technologies have been studied for the reduction of nuclear waste burden. One option for the alternative is nuclide management by the chlorination and separation of alkaline earth, TRU (TransUranium), and rare earth elements after the thermal treatment of used nuclear fuel. The resulting products are uranium oxide and AE/TRU/RE phosphate blocks. In this study, it was investigated on the evaporation of the alkaline earth and some rare earth chlorides in the vacuum distillator. The salt evaporation experiments were carried out at various temperatures. The vapor pressure of chloride varies widely depending on the type of chloride, and it is needed to investigate the evaporation behavior of individual chloride. In case of BaCl_2 and SrCl_2 , the separation with engineering significance by evaporation was possible from about $1,070^\circ\text{C}$. In case of LaCl_2 and CeCl_2 , the separation was possible from about 900°C . But it was found that rare earth chloride had a more problematic side reaction at high temperatures from XRD analysis. It is necessary to develop an appropriate method for the evaporating and separating all chloride at once.

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Keywords: Alkaline earth element, Rare earth element, Chloride, Vacuum distillation

Observations for the SF Volume Reduction Using SNF Rod Consolidation

Beomgyu Kim*, Kyungho Roh, Kiyoun Kim, and Taehyung Na

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kbbk08912@khnp.co.kr

As spent nuclear fuel in overseas nuclear power plants began to fill up pools in the mid-to-late 1980s, more than ten reports and patents were published between the late 1980s and early 1990s to reduce SF volume. In this study, it implemented qualitative analysis on the rod consolidation possibility to reduce SF volume affecting the spent fuel safety.

Densification technology was a technique that attracted attention for improving storage and disposal efficiency at US, Swedish, German, French, and Korea during the 1980s and 1990s. Each country conducted various studies considering different structures of nuclear fuel rods, with particular interest shown by the United States and Germany. Among them, the United States even carried out demonstration experiments.

In the case of domestics, there are various design variables in dense storage operations such as dry or wet packaging environments for fuel rods, extraction methods of fuel rods (horizontal vs vertical), and withdrawal sequences (one by one, row by row, or all at once). Since 1989, NUS has been selected as a contractor for Phase III of the Department of Energy's (DOE) Plutonium Cladding Development Project (PCDP).

In this study, it observed some considerations on rod consolidation that is likely to improve SF volume reduction based on NUS's Phase II design data. The dense process device building is separated into contaminated and clean zones. it revealed that there were issues needed to progress rod consolidation in terms of fuel assembly insertion such as inserting fuel assemblies in the required amount at storage racks inside the process, visual inspection and serial number reading for fuel assemblies and positioning fuel assemblies horizontally or vertically. Additionally, it showed that some precautions as dealing with fuel rod consolidation when extracting fuel rods from assemblies and rearranging them into compact bundles such as securing vertical position of assemblies to extract one column of fuel rods at once, extracting fuel rods one column at a time using grippers attached to the ends of the rods and transferring extracted fuel rods sideways and depositing them into the proper quantity hoppers for reassembly, Consolidated nuclear fuel being inserted into canisters with pusher.

Using SNF rods to reduce volume offers several advantages as follows. 1. Space savings: Reducing the volume of nuclear fuel rods while optimizing storage space can improve operational efficiency in nuclear power plants. 2. Safety improvements: Increasing the density within the fuel rods reduces the risk of radioactive material leaking into the environment, which helps minimize environmental damage that may occur during accidents. 3. Cost reduction: Lowering the costs associated with handling and disposal of fuel rods ultimately leads to reduced overall expenses for producing nuclear energy. 4. Resource conservation: Recycling spent nuclear fuel and converting it back into new fuel allows for more efficient use of precious uranium resources, slowing down the rate at which natural resources deplete. 5. Sustainability enhancement: Addressing challenges related to waste management contributes to ensuring future generations have access to clean and safe energy sources. Based on this study, we believe that partial utilization of low-burnup spent nuclear fuel is efficient for current dense storage technology as long as removal of high heat generating nuclides is not required. However, further research is needed to determine quantitatively how much storage efficiency can be improved through additional study.

Keywords: Spent nuclear fuel, SF volume reduction, Rod consolidation

A Review Study on Oxidation Behavior of UO_2 at Low Temperature

JangSoo Oh* and Dong-Hak Kook

Korea Atomic Energy Research Institute (KAERI)

*jangsoo@kaeri.re.kr

The gradual oxidation of UO_2 in air, resulting in the formation of U_3O_7 / U_4O_9 and U_3O_8 , has been the subject of extensive research for decades. This is due to the significance of the process for the dry storage and disposal of spent nuclear fuel, as well as for the storage for UO_2 powder and certain fuel-recycling processes.

The primary issue regarding fuel oxidation in terms of storage and disposal of spent nuclear fuel is the generation of U_3O_8 or other high volume phases. These may result in cladding failure and fuel powdering, which present significant challenges in ensuring the safe storage and disposal of spent nuclear fuel. A number of important element of the oxidation process remain inadequately explained. Of particular interest is the pronounced discrepancy in the oxidation behavior of unirradiated UO_2 in comparison to that observed in spent nuclear fuel, which has yet to be fully explained.

In this study, the oxidation of UO_2 at temperatures below 400°C was analyzed in accordance with the AECL in order to obtain a deeper understanding of the oxidation of UO_2 . The primary parameters influencing the oxidation rate of UO_2 were investigated. Also, the oxidation mechanisms, oxidation rate, and discrepancies in the oxidation behavior of unirradiated UO_2 and spent nuclear fuel were examined

Acknowledgements

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Keywords: UO_2 , U_3O_8 , Oxidation, Dry storage, Disposal of used fuel

Development of Grinder/Polisher for Hot Cell Application

Jae-Han Kim

Korea Atomic Energy Research Institute (KAERI)

jaehankim@kaeri.re.kr

Metallographic analysis is a critical process for examining the microstructural properties of materials, typically requiring the use of a grinder/polisher to prepare specimens. This preparation process involves mounting the specimen, followed by grinding and polishing to achieve a surface suitable for optical analysis. However, applying this process to radioactive materials presents significant challenges. The potential for radiation exposure poses risks to human operators, and the electronic components within standard grinder/polisher equipment may fail when exposed to radiation. To address these challenges, we have designed and developed a new grinder/polisher system specifically for use in radioactive environments. This system is designed without electronic components in the main part of the machine, with the control unit installed outside of the hot cell, thus eliminating the risk of radiation-induced failures. Furthermore, it is optimized for operation within a hot cell, a concrete-shielded environment designed to safely contain radioactive materials. The system is also easily manipulated via robotic arms, ensuring precise and safe operation in hazardous conditions. Our development provides an effective solution for specimen preparation in the field of metallographic analysis of radioactive materials, enabling safer and more reliable results.

Keywords: Grinder/polisher, Hot cell, Metallographic analysis, Radioactive materials

Johnson-Cook Model for Canister Material in Spent Nuclear Fuel Storage Container

Yunyoung Yang*, Jongmin Lim, and Sang Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*yang0124@kaeri.re.kr

This research paper focuses on the development of the Johnson-Cook model considering into high strain rate properties of the base material and SA (Submerged-Arc)-welded specimens of stainless steel 304, which is used for the canister in spent nuclear fuel storage containers. The primary objective of this study is to enhance the understanding of the mechanical behavior of these materials under various strain rates, which is critical for ensuring the structural integrity of the canister under different operational conditions. The Johnson-Cook model employed in this study consists of the various material constants representing the strain hardening, strain rate hardening, and temperature. For the strain term, A was determined using the yield stress at a strain rate of 0.001 /s, B represents the strain factor, and n is the strain hardening exponent. For the strain rate term, C is the strain rate hardening factor. Those material constants are determined for the strain and strain rate terms using tensile tests conducted on two types of specimens: the base material specimen in the rolling direction of stainless steel 304 and the SA-welded specimen of the same material. The tests were performed at six different strain rates: 0.001, 0.01, 0.1, 1, 10, and 100 /s. The results showed that the Johnson-Cook model could accurately predict the behavior of both the stainless steel 304 base material and the SA-welded specimens. These findings provide valuable insights into the mechanical properties of canister materials used in spent nuclear fuel storage, potentially informing the design and safety assessments of such containers. The study contributes to a broader understanding of material behavior under high strain rates, which is crucial for applications in nuclear energy and other industries requiring high-performance materials.

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Keywords: Johnson-cook model, Spent nuclear fuel storage, Canister, High strain rate, Submerged arc weld

Insight for the Thermal Evaluation Licensing Review of MAGNASTOR Using U.S. Licensing Experience

Daesik Yook

Korea Institute of Nuclear Safety (KINS)

dsyook@kins.re.kr

The introduction of the design approval for spent fuel storage cask into the Nuclear Safety Act in 2021 led KHNP to select a canister-based spent fuel storage cask as the dry storage system for use at light water reactor nuclear power plant sites. This chosen storage system closely resembles the MAGNASTOR system by NAC, which has been granted a Certificate of Compliance by the U.S. NRC. The aim of this study is to analyze the NRC's review experience to identify potential items that should be preemptively reviewed when considering domestic licenses.

The NRC's thermal evaluation primarily focused on assessing normal, abnormal, and accident conditions to ensure that the temperature of the cladding remains below critical limits during normal storage. This is crucial to prevent a gross rupture that could lead to cladding failure. The thermal analysis process involved verifying the similarity between the actual design geometry and the geometry used in thermal simulations, as well as considering the physical effects that design features may have on thermal performance.

The MAGNASTOR system, which features four inlets at the bottom and 26 outlets at the top, prompted the NRC to conduct a detailed review due to concerns that these system characteristics could result in low-speed winds, potentially degrading thermal performance. It was acknowledged that the peak cladding temperature (PCT) could be underestimated due to discrepancies between the actual steady-state geometry and the assumptions used in the thermal analysis, the analysis model itself, and the wind speed conditions used in the initial environmental boundary conditions. However, the NRC ultimately concluded, based on its own study (NUREG-2174), that even if there were flaws in the applicant's assessment, the actual temperature rise in the licensed system would not result in a PCT exceeding the allowable limit.

In contrast, Korea has not yet established regulatory positions through self-studies conducted by regulatory bodies, as the NRC has done. Therefore, applicants will need to demonstrate to domestic regulators that the maximum cladding temperature is not underestimated. This will require detailed analyses that take into account the similarity between the actual geometry and the thermal analysis model, the evaluation of low-speed winds when used in a building, and the initial ambient boundary temperatures corresponding to the domestic storage environment.

Keywords: MAGNASTOR, Thermal evaluation, Low-speed wind effect, PCT temperature

Structural Integrity Assessment of Cask Test Equipment for Verification of Thermal Characteristics Computational Model

JaeYong Kim*, YongSik Yang, DaeHo Kim, and DongHak Kook

Korea Atomic Energy Research Institute (KAERI)

*kjdkjy@kaeri.re.kr

In order to verify the thermal characteristics computational model of the dry storage system of the spent fuels (SFs) of a pressurized heavy-water reactor, a cask test equipment with an insulated condition will be manufactured. This test equipment has a structure similar to a silo, a dry storage system of pressurized heavy-water reactor SFs, with nine baskets stacked inside, but is axially segmented to simulate various test conditions and install numerous thermocouples. In addition, in order to simulate the decay heat of SF bundles of a pressurized heavy-water reactor, which is harmful to human and can only be handled in a special test facility (such as a hot cell facility), the nuclear fuel assembly bundles are replaced with heater. Moving the basket with the heater in the axial direction and adjusting the power of the heater inside the basket will simulate various temperature fields not only in the axial direction of the basket but also in the radial direction. Such various test conditions and temperature fields will be utilized in verifying the developed thermal characteristics computational model.

In this paper, the procedures and results of the structural integrity evaluation of the cask test equipment are described. In order to evaluate the integrity of a silo structure installed in the field, static analysis, thermal stress analysis, drop impact analysis, and seismic analysis must be performed. However, because this test equipment will be installed inside a facility that is not affected by the environment, and the test equipment is planned to be operated while being fixed to the internal structures of surrounding facilities using eye bolts, drop impact analysis and seismic analysis were judged unnecessary and were excluded from the structural integrity evaluation items. Since the weight of one basket containing a total of 60 pressurized heavy-water reactor SF bundles is approximately 2 tons, a weight of approximately 16 tons is imposed to the 9th basket at the very bottom. In addition, a thermal load is applied due to the temperature difference caused by the difference in the heater power and location of the heater installed inside the basket. In this way, the integrity of the basket structure due to external static load and thermal load was analyzed using ANSYS, a commercial finite element analysis software, and stress linearization was performed to compare it with the stress limits presented in ASME section-III, subsection NB for the structural integrity assessment.

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Keywords: Structural integrity assessment, Cask test equipment, Pressurized heavy-water reactor, Static analysis, Thermal stress analysis, ASME

Conceptual Design of Spent Nuclear Fuel Interim Storage Facility

Hyungoo Kang*, Changmin Shin, Sanghwan Lee, and Taechul Moon

Korea Radioactive Waste Agency (KORAD)

*kanghyungoo@korad.or.kr

This study outlines the conceptual design of storage facilities necessary for the safe management of spent nuclear fuel generated from nuclear power plant. First, by analyzing the operational cases of PWR nuclear spent fuel interim storage in leading countries, the study identified the need for temporary storage locations within the facility, the necessity for sufficient structural integrity, and the requirement for single-failure accident prevention functions. Based on these findings, this study proposed operational concepts and required capacities for facilities suited to domestic conditions, presenting conceptual design plans for two types of interim storage methods: in-door and out-door facilities.

The site and operational scale estimates for the interim storage facilities were based on conservative assumptions, considering the maximum capacity of each storage facility. Therefore, the operational concepts for interim storage facilities derived from this study may need to be adjusted in response to fluctuations in PWR spent nuclear fuel generation or changes in domestic policy conditions. Considering this, it is suggested that the results of this study be actively utilized in the national-level advancement of interim storage facility projects for PWR spent nuclear fuel. Furthermore, it is anticipated that subsequent research will be necessary to develop more rational operational strategies.

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Keywords: Spent nuclear fuel, Interim storage facility, Interim storage operational concept

Preliminary FMEA Application for HVAC Systems in Nuclear Fuel Cycle Facilities

Woojin Jo*, Hyojik Lee, Sang-jin Park, and Seungnam Yu

Korea Atomic Energy Research Institute (KAERI)

*wjjo@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI) conducts various research and development projects related to nuclear energy. Among them, nuclear fuel cycle research is essential for the sustainable utilization of nuclear power systems, and KAERI actively engages in diverse studies in this field. Such studies requires facilities, known as nuclear fuel cycle facilities, which are equipped with hot cells capable of handling irradiated materials and spent nuclear fuel. The hot cells in these facilities require exceptionally high safety management standards because they handle radioactive materials with high levels of radiation through remote operations, as these materials are difficult for humans to access. In particular, to prevent the leakage of radioactive substances like nuclear materials, HVAC systems operate continuously 24/7, maintaining negative pressure within the hot cells. Therefore, HVAC systems in nuclear fuel cycle facilities are crucial; they require thorough maintenance, management, and safety validation through assessments starting from the design stage.

Various methods exist for safety evaluation, but this study focuses on Failure Modes and Effects Analysis (FMEA). FMEA is a systematic analytical method that identifies potential failure modes of system components and assesses their possible consequences, thereby contributing to the overall safety evaluation. It is a widely used and well-established qualitative reliability assessment methodology in the field of reliability engineering.

KAERI operates two nuclear fuel cycle facilities: the Irradiated Material Examination Facility (IMEF) and the Post-Irradiation Examination Facility (PIEF). Although each facility comprises various systems, this study specifically focuses on performing FMEA on the HVAC systems, which pose the most significant potential risk to the safety of workers and the public. The results of this study provide a qualitative understanding of the extent to which HVAC system components contribute to system failures, enabling the prioritization of maintenance and safety management efforts. Furthermore, these findings can be utilized as input data for condition-based maintenance strategies, aiding in the safe management of nuclear fuel cycle facilities.

Acknowledgements

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Keywords: FMEA, HVAC, Nuclear fuel cycle facility, Safety

Solvent Extraction Behavior of Europium, Uranium, and Neptunium in 30% Tributyl Phosphate/n-Dodecane

Seungmo Yeon¹, Seongye Kwon², Namcheol Kim^{3,4}, Dong Hun Choi⁵, Yongheum Jo⁶, Kwangheon Park⁵,

Won Pyo Jeong⁷, Seungyeon Choi⁷, and Jaeyeong Park^{2,*}

¹Korea Advanced Institute of Science and Technology (KAIST)

²Ulsan National Institute of Science and Technology (UNIST)

³Korea Atomic Energy Research Institute (KAERI)

⁴University of Science and Technology (UST)

⁵Kyung Hee University

⁶Hanyang University

⁷Korea Nuclear International Cooperation Foundation (KONICOF)

*jypark@unist.ac.kr

The solvent extraction method using tributyl phosphate (TBP), known as the PUREX process, is the most commercially successful reprocessing method for spent nuclear fuel in the back-end of the nuclear fuel cycle. Understanding the aqueous chemistry of actinide elements contained in spent nuclear fuel is critical for the efficient and stable operation of the PUREX process. Neptunium-237 (²³⁷Np), which constitutes about 0.05% of spent nuclear fuel, is one of the long-lived actinide isotopes. Recovering ²³⁷Np alongside uranium can help mitigate the long-term storage burden of high-level waste generated in reprocessing.

In this experiment, the distribution coefficients of europium (Eu), uranium (U), and neptunium (Np) were measured between an aqueous nitric acid solution and a 30% TBP/n-dodecane solution. Aqueous samples containing Eu, U, and Np were prepared with three different nitric acid concentrations (0.100 M, 0.802 M, and 3.922 M). The concentrations of Eu, U, and Np were the same across all samples at 2.00×10^{-7} M, 3.90×10^{-2} M, and 2.39×10^{-12} M, respectively. For solvent extraction, 2 mL of the aqueous phase and 2 mL of the 30% TBP/n-dodecane organic phase were mixed in PTFE Oak Ridge-style tubes and agitated for 30 minutes. ¹⁵²Eu and ²³⁹Np isotopes were used for Eu and Np, with concentrations analyzed via gamma counting using high-purity germanium (HPGe) detectors. Uranium, with its natural isotopic ratio, was analyzed using a photo/electron-rejecting alpha liquid scintillation (PERALS) spectrometer (model 8100AB, ORDELA INC.) for gross alpha measurements.

After phase separation, the distribution coefficients (D) of each element were calculated based on concentrations of the aqueous and organic phases. In the plot of the distribution coefficient against nitric acid concentration, it was observed that the log D values for Eu and U increased linearly with increasing log [HNO₃], whereas the log D for Np showed a nonlinear increase with log [HNO₃]. This behavior indicates that the dominant chemical species of Eu and U remained as Eu³⁺ and UO₂²⁺ across all three nitric acid concentrations, while for Np, the redox potential (E_h) of the solution, influenced by dissolved oxygen, increased with rising nitric acid concentration, leading to a higher proportion of NpO₂²⁺ compared to NpO₂⁺. The changes in the dominant Np species in each sample were confirmed using an E_h -pH diagram generated with HSC Chemistry 10.0 software (Outokompu Research Oy).

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Keywords: Solvent extraction, Europium, Uranium, Neptunium, Tributyl Phosphate (TBP), PUREX

Advanced Chlorination Techniques for Reducing High-Level Waste Volumes in Nuclear Fuel Management

Dalsung Yoon*, Sangkwon Lee, Ju Ho Lee, Eunsoo Lee, Seungwoo Paek, and Changhwa Lee

Korea Atomic Energy Research Institute (KAERI)

*dyoon@kaeri.re.kr

With temporary storage facilities for spent nuclear fuel (SNF) nearing saturation, the urgency to develop effective treatment solutions for SNF is becoming increasingly critical. Direct disposal of SNF is a commonly employed method; however, this approach categorizes the entire fuel assembly as high-level waste, leading to significant challenges and expenses associated with disposal. To mitigate these issues and enhance the safety of long-term storage, the advancement of SNF treatment technologies is paramount. Such advancements also necessitate continuous and concerted efforts from a national policy perspective to ensure their successful implementation and sustainability.

This study specifically focuses on minimizing the volume of high-level waste generated from light water reactor fuel by separating uranium, which constitutes the majority of SNF. By achieving this separation, the overall disposal burden can be significantly reduced. To explore this approach, we conducted a series of lab-scale experiments to investigate the chlorination characteristics of both uranium and rare earth (RE) oxides. These experiments employed NH_4Cl as the chlorinating agent, and the subsequent assessment evaluated the feasibility of uranium separation from a simulated U/RE/Sr fuel mix.

The results demonstrated promising avenues for the efficient separation of uranium from other components in the SNF, thereby potentially decreasing the volume of high-level waste. Additionally, the study provides insights into the chemical behavior of uranium, RE, and strontium oxides under chlorination conditions, contributing valuable data to the development of more effective SNF treatment technologies. Comprehensive findings and detailed analyses will be discussed in forthcoming publications and presentations, shedding light on the methods, outcomes, and implications for future research and policy development in SNF management.

Keywords: Spent nuclear fuel, Uranium, TRU, NH_4Cl , Chlorination

Corrosion Behavior Analysis of Internal Structure Materials (Al6061 and SUS 17-4 PH) in a Dry Cask Environment

Kwanil Chae, Seongkoo Hong, Samuel Park, Sangjin Kim, and Sungyeol Choi*

Seoul National University (SNU)

*chosys7@snu.ac.kr

Recently, the U.S. Nuclear Regulatory Commission (NRC) has raised concerns regarding the residual moisture remaining inside dry storage casks after vacuum drying. This residual moisture is believed to significantly impact the degradation of internal structures. The presence of residual moisture affects the degradation of various internal components in the following ways: (i) Oxidizing agent and (ii) Radiolysis-induced radicals. Therefore, it is crucial to evaluate the degradation behavior of internal structures considering the presence of residual moisture. However, the corrosion behavior of AA6061 and SUS17-4, which are used as internal structural materials in the domestically developed KORAD-21 transportation and storage casks, has not been evaluated.

In this study, the corrosion behavior was evaluated electrochemically, taking into account the internal environment of dry storage casks. The internal environment of dry storage casks includes the following factors: (i) pH conditions from 4 to 6 were selected, based on studies of pool composition changes when nuclear fuel is immersed; (ii) hydrogen peroxide (H_2O_2) concentrations from 10^{-6} M to 10^{-4} M, generated by the radiolysis of residual moisture, were considered; and (iii) conditions fully saturated with oxygen.

The results of the electrochemical analysis are as follows: (i) The corrosion potential exhibited a tendency to increase with a decrease in pH. (ii) Similarly, an increase in the concentration of H_2O_2 also correlated with an increase in the potential. Consequently, it appears that AA6061 and SUS 17-4 PH materials used in dry storage casks may experience significant corrosion effects in the internal environment. This result leads to the following conclusions: (i) The anticipated impact of the internal environment on corrosion includes the potential influence of H_2O_2 acting as an oxidizing agent. (ii) This factor may also affect the corrosion rate.

Based on the results of this study, the findings are expected to enable a more conservative evaluation of internal container conditions. Ultimately, this will provide a basis for assessing the corrosion integrity of dry storage containers.

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Keywords: Spent nuclear fuel, Dry storage, Corrosion, Hydrogen peroxide (H_2O_2), Residual moisture, Gamma radiation, Neutron exposure, Al6061, SUS 17-4 PH

Leaching Stability Assessment of Uranium Oxide Blocks Fabricated via Slip Casting Process

Su Jeong Heo*, Kweonho Kang, Jung-Hoon Choi, Ki-Rak Lee, and Changhwa Lee

Korea Atomic Energy Research Institute (KAERI)

*sujeongheo@kaeri.re.kr

Spent Fuel (SF) storage is a critical global issue for utility companies. Delays in securing disposal sites and storage facilities increase economic costs and environmental risks due to the need for additional storage space. Developing solidification technology for uranium oxide blocks is essential for the safe disposal of research-related SF. This study explored the fabrication of uranium oxide (UO₂) blocks using depleted uranium (DU) powder through a slip casting process, typically used in pottery. The process was refined across six key steps: binder preparation, sludge preparation, injection/molding, demolding, wet drying, and sintering, leading to the successful production of blocks with diameters of 30 and 60 mm. To evaluate the long-term stability of these blocks, we conducted leaching tests following ASTM C1308-21 standards. Each pellet was placed in a vessel filled with deionized (DI) water at ambient temperature. The test solution was periodically removed and replaced with fresh water at specific intervals, and the vessel was returned to ambient conditions. The leachates were analyzed using ICP-MS to measure the extent of UO₂ leaching. These tests are crucial for assessing the durability and safety of the uranium oxide blocks as a potential technology for spent nuclear fuel management. Future assessments, including mechanical tests and detailed leaching analysis, will further validate this method.

Keywords: Uranium oxide blocks, Spent fuel management, Slip casting process, Leaching tests, Depleted Uranium (DU)

Dissolution Behaviors of BaO Into Molten Salts and Separation of Dissolved Ba With Precipitants

Seol Kim^{1,2}, Chang Hwa Lee², and Eun-Young Choi^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*eychoi@kaeri.re.kr

The utilization of molten salt to separate highly radioactive nuclides from used nuclear fuel and subsequently precipitate them into solids is garnering attention as a means to reduce the disposal area of used nuclear fuel. Our previous work has demonstrated the efficient separation of Sr, a highly radioactive nuclide, from simulated nuclear fuel using molten MgCl_2 and LiCl-CaCl_2 salts. Ba, although not highly radioactive like Sr, shares similar characteristics as a group 2 alkaline earth metal element. Therefore, if Sr can be effectively separated using molten salts, it is likely that Ba can be as well. However, the behavior of Ba in the molten salts utilized for Sr separation remains unexplored.

In this study, we examine the reaction of BaO with various salts (i.e., LiCl , LiCl-CaCl_2 , CaCl_2 , and MgCl_2) by immersing BaO reagent in the respective salts and analyzing both the salt and the residual solid. It was initially assumed that the behavior of Sr and Ba in various molten salts would be similar, but we discovered that the behavior of Ba and Sr in these salts is different. Sr primarily exists as an oxychloride in various salts, while Ba is present either as a chloride or in a form that cannot be identified by X-ray diffraction (XRD). Additionally, while Sr could be successfully precipitated using the precipitant K_2CO_3 , Ba could not be precipitated using the same method. Instead, it was found that Ba could be effectively precipitated as a solid when using MgSO_4 as a precipitant. We are currently exploring precipitants that can effectively precipitate both nuclides under various conditions, aiming to develop a method for efficiently separating radioactive nuclides from used nuclear fuel.

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Keywords: Molten salt, Separation, Barium, Used nuclear fuel, Precipitation

Conceptual Framework for Automation of Electrorecovery Unit Process in Pyroprocess Automation Mockup

Jonghui Han*, Jong Kwang Lee, Byung Suk Park, and Seungnam Yu

Korea Atomic Energy Research Institute (KAERI)

*jhan@kaeri.re.kr

To achieve the automation of pyroprocessing operations, both the automation of the equipment itself and the automation of the system for handling and transporting process materials are necessary. Each piece of process equipment and handling device needs specific controllers for operational conditions, as well as a higher-level controller to coordinate the overall operation sequence. Previous studies proposed the pyroprocessing execution system and presented a preliminary framework concept for its implementation. Subsequent studies introduced a modified framework that includes the maintenance concept of installed process equipment in hot cells.

Based on the preliminary framework concept proposed in previous studies, this research presents an automation operation framework for the electrorecovery process to be installed in the pyroprocessing automation mockup. This unit process comprises electrorecovery equipment, electrorecovery distillation equipment, and a device for basket assembly/disassembly. In the mockup, a gantry and handling tools are used for transporting process baskets automatically. The proposed framework includes hardware and software configurations, operation procedures and definitions of events for coordinated control of process equipment and handling devices.

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Keywords: Pyroprocessing, Electrorecovery process, Process automation, Mockup

Design of Simulated Test Specimens to Verify the Integrity of CANDU Spent Fuel Cladding During Long-Term Dry Storage

Joonho Moon*, Jong-Dae Hong, Hyun-Gil Kim, and Donghak Kook

Korea Atomic Energy Research Institute (KAERI)

*jhm111@kaeri.re.kr

As the duration of dry storage of spent nuclear fuel continues to increase globally, there is a growing interest in the long-term stability of dry storage systems. In South Korea, dry storage of CANDU spent fuel (CSF) has been operational for about 30 years. However, there is currently no experience in confirming the degradation behavior of spent fuel stored in dry storage. In order to demonstrate the safety of a dry storage system, it is very important to predict the degradation behavior of the spent nuclear fuel cladding, but there are many difficulties in conducting experiments using high-level radioactive materials. Therefore, simulated test specimens were designed with the characteristics of CSF cladding to verify the corrosion and degradation behavior of the material during long-term dry storage.

The material of the simulated test specimen is zircaloy-4, with an outer diameter of 13 mm and a thickness of 0.4 mm. In a CANDU reactor, an average burnup is about 7.8 GWD/MTU. According to previous literature, the oxidation thickness of the CSF cladding with a burnup of 7.8 GWD/MTU is 3 to 4 μm , and after 9 years in dry storage, the oxidation thickness averages about 6 μm , indicating that most of the corrosion occurred before dry storage. However, in this case, it should be considered that oxidation may have been limited in the dry storage due to low-oxygen atmosphere. More oxidation may occur in the air, which is the CSF dry storage environment. In consideration of this, it aims to grow an oxide thickness of 10 μm or more in zircaloy-4 specimens. However, it is known that an oxide thickness of about 0.23 μm is formed in 100 years and 2 μm in 1,000 years in an air at 150°C, which is a dry storage condition. Therefore, because the experiment takes too much time under the actual dry storage, it is necessary to select acceleration conditions. If the temperature rises above 500°C during oxidation, the oxidation behavior may change, so it should be considered. According to the NUREG/CR-6846 report, there is a possibility that the oxide thickness will grow up to 10 μm when oxidation is performed at 400°C for 1,000 hours in an air, so the experiment will be conducted based on this. Additionally, the residual ductility will be measured by conducting a Post-Quench Ductility experiment using a ring compression test.

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Keywords: PHWR, CANDU spent fuel, Dry storage, Oxidation, Zircaloy-4

Establishment of Test Plan for Evaluating Radiation Shielding Performance Under Fire Exposure of Concrete

Siwan Noh*, JaeHoon Lim, and Sang Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*snoh@kaeri.re.kr

Concrete is widely used for radiation shielding because it has adequate radiation shielding ability for both photons and neutrons, is easy to construct, is inexpensive, and can be used as a structural material. It is known that when concrete is exposed to high temperatures due to a fire, the moisture is mainly reduced, resulting in a decrease in density, and cracking due to high temperature may also occur. In particular, when moisture is reduced, hydrogen, which is most effective for neutron moderation, is reduced, which can significantly reduce the neutron shielding performance. Therefore, the shielding performance test for gamma rays and neutrons after exposing actual concrete specimens to fire will be conducted to determine the level of shielding capacity loss, and a test plan was developed for this study. The concrete specimens have been manufactured and will be tested for shielding ability against gamma rays and neutrons before exposure to fire. The same specimens are then exposed to fire and the same shielding tests are performed to evaluate the level of shielding degradation before and after the fire exposure. The shielding performance test for gamma rays and neutrons is performed using Cs-137 and Cf-252 sources at the KAERI's irradiation facility, and the MCNP computer program is used to calculate the expected shielding capability and design the collimator in advance. Based on the test plan established in this study, a radiation shielding performance test for fire exposure of concrete will be conducted next year.

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Keywords: Concrete, Radiation shielding, Fire, Moisture, Neutron, Gamma rays, MCNP

Comparison of MCNP Source Definition on SNF Analysis

Seung Uk Yoo*, Sung Hee Park, and Moonoh Kim

KONES Corp.

*yooswok@kones21.com

In recent, various research have been conducted on methods for analyzing Spent Nuclear Fuel (SNF). Specifically, there is ongoing discussion regarding the most accurate method for analyzing inventories of Special Nuclear Material (SNM) within SNF using computational codes. Generally, the inventories of SNM in SNF has been calculated using ORIGEN, a source term analysis code, and the neutron spectrum and detector analysis have been conducted using MCNP, which applies Monte Carlo analysis techniques.

In the conventional SNF analysis, since the calculations rely on neutrons, the particle type in the source term card is defined as a neutron in MCNP, and the neutron spectrum from ORIGEN is used as an input. However, if the particle type in MCNP is defined as Spontaneous Fission (SF) at source definition card, neutron spectrum from ORIGEN is no longer required, and only requires the composition of fissile material in material card. Using SF particle in MCNP could simplify the process of SNF analysis. Thus, analyses were conducted for figuring out the difference between two different approaches.

The composition of the SNF was calculated using ORIGEN for WH 17×17 fuel with burnup of 55,000 MWd/MTU and cooling time of 1 year. The analyses were conducted on a single fuel rod with and without coolant, a 3×3 fuel assembly, and a 17×17 fuel assembly with coolant. The calculation results showed that the input using SF as the particle type (SF model) and the input model using the neutron spectrum derived from ORIGEN (ORIGEN model) produced similar neutron spectrum in the thermal region, however, differences were observed in the fast region, due to the neutron spectrum derived from ORIGEN. On the other hand, when a He-3 detector was simply simulated at a distance of 10 cm from the assembly to compare the total proton generation, the SF model produced a slightly higher total number of proton than the ORIGEN model. This is likely due to the fact that He-3 neutron absorption reactions occur primarily in the thermal region.

Additional analysis will be conducted for various composition of SNF to figure out the detailed differences between SF model and ORIGEN model.

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Keywords: MCNP, ORIGEN, Spent nuclear fuel, Spontaneous fission

Validation of Nuclear Fuel Prediction for Application of Initial Characteristics Data of Spent Nuclear Fuel

Kyung-Tae Kim*, So-Young Kim, Oh-Hyun Kwon, and Seong-Ki Lee

KEPCO Nuclear Fuel (KEPCO NF)

*kkim@knfc.co.kr

Before the operation of the dry storage facility for spent nuclear fuel, acceptance criteria for interim storage must be established promptly, and to achieve this, initial characteristics data for the spent nuclear fuel must be identified.

To assess the initial characteristics of spent nuclear fuel based on Zr-alloy cladding, and investigation was conducted on the pool side examination (PSE) and post irradiation examination (PIE) data for spent fuel in wet storage pools. Through this study, a comparative analysis was performed between the measured and the prediction performance data to verify the adequacy of the initial characteristic data.

As a result, it was confirmed that the values predicted by the approved fuel rod design code can be used as initial characteristic data for spent nuclear fuel.

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Keywords: Spent nuclear fuel, Zirconium-alloy, Initial characteristic data

Validation of Transient CFD Simulation Model of Vacuum Drying in Dry Cask Simulator

Doyun Kim* and Seunghwan Yu

Korea Atomic Energy Research Institute (KAERI)

*dykim0129@kaeri.re.kr

In order to investigate the transient temperature of spent nuclear fuel assembly inside a dry storage system, Transient CFD simulation model was developed to cover the vacuum drying condition. Dry cask simulator was used to validate the developed CFD transient model. Dry cask simulator measured the temperature the fuel assembly and thermocouple lance in a vacuum condition to analyze TC lance temperature anomaly. Initial temperature was identical to the atmospheric temperature, but after heaters were turned on, the temperatures started to increase. The temperature increase transient under vacuum condition was measured in the dry cask simulator, and the test result were used to validate the CFD transient model. Transient time step of 3,600 second and 100 iteration per time step was used. CFD results underestimated the measured temperature, but the developed CFD mode using both of explicit and porous model showed allowable agreement with the experimental data during 24 hours of temperature transient due to the onset of heating.

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Keywords: Dry storage, Vacuum drying, CFD simulation, Transient simulation, Dry cask simulator

Fracture Assessment of Semi-Elliptical Defects in Geometrically Discontinuous Welded Joints of Spent Nuclear Fuel Storage Canisters

Jongmin Lim* and Sang Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*jmlim86@kaeri.re.kr

In this study, fracture assessments were performed on three-dimensional semi-elliptical defects located in geometrical discontinuities of welded joints in spent nuclear fuel storage canisters. The storage canister is inspected for defects during manufacturing and preservice, in-service inspections. However, design codes, such as ASME BPVC XI, provide allowable defect sizes and evaluation methods for components without geometrical discontinuities, such as pipes or cylinders. No specific guidelines exist for welds at geometrical discontinuities, such as lid-to-cylinder or bottom-to-cylinder joints.

Geometrical discontinuities cause stress concentration, leading to higher stress levels, which increase the possibility of crack initiation and propagation. Defects in structures can result in brittle fracture if critical fracture toughness is exceeded or plastic collapse if the load exceeds the limit. The Failure Assessment Diagram (FAD)-based fracture assessment method was employed in this study to represent unstable fracture and plastic collapse modes. The acceptability of defects was determined based on the position of the failure assessment points on the FAD.

Linear Elastic Fracture Mechanics (LEFM)-based Finite Element Method (FEM) analysis was conducted to calculate stress intensity factors for the defects. Additionally, non-linear analysis incorporating perfectly-plastic properties was performed to determine the limit load due to the defects. Comparative fracture assessments were carried out for geometrically discontinuous bottom-to-cylinder welded joints and geometrically continuous cylinder-to-cylinder welded joints. These results were represented on the FAD and analyzed for comparison.

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Keywords: Spent nuclear fuel storage canister, Fracture assessment, Failure Assessment Diagram (FAD), Stress intensity factor, Limit load

Application of Submodeling Technique for Fracture Analysis of Spent Nuclear Fuel Storage Casks Under Impact Conditions

Jongmin Lim* and Sang Soon Cho

Korea Atomic Energy Research Institute (KAERI)

*jmlim86@kaeri.re.kr

To perform fracture analysis through the finite element method (FEM), the use of highly fined and detailed meshes is required to accurately reflect the geometrical characteristics of defects. In particular, for accurate fracture analysis of semi-elliptical defects, commonly used for defect definition, spider web meshing is typically employed. This approach demands a large number of nodes and elements, significantly increasing the complexity of the analysis.

Moreover, when explicit time integration is applied, the highly fined mesh leads to an excessively small minimum time increment, causing the analysis to become computationally infeasible. To address these challenges, this study employed the submodeling technique to evaluate the fracture behavior of a canister in spent nuclear fuel storage canisters under impact conditions.

The submodeling technique separates the global structure (global model) and the region of interest (local model), applying different analysis methods to each. This approach enables detailed analysis in critical areas while conserving computational resources. It is primarily used for complex analyses that require consideration of both the global model and localized fine-detail analysis.

The application of the submodeling technique in fracture analysis under impact conditions for spent nuclear fuel storage containers offers significant advantages. First, computational efficiency is improved, as defect modeling in explicit analysis often results in a reduced time increment, which greatly increases computational cost. By employing submodeling, explicit time integration is utilized for the global model, while implicit time integration is applied to the local analysis, thus reducing computation time. Additionally, this technique ensures accuracy in detailed analysis by allowing explicit methods to handle complex dynamic phenomena, such as impacts, in the global model, while implicit methods are used for more precise fracture analysis of defects in the local model, enabling thorough evaluation in critical areas.

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Keywords: Spent nuclear fuel storage, Impact analysis, Fracture analysis, Submodeling technique, Flaw modeling

Preliminary Evaluation of Criticality Safety Due to Boron Depletion in Neutron Absorbers of a Dry Storage System Using SCALE 6.2.4

Ji Su Kang and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Subcriticality of spent fuel in dry storage systems (DSSs) can be maintained by placing neutron absorbers or poison plates around the fuel assemblies. Commonly used neutron poison materials include borated stainless steel, boron-containing aluminum alloys, aluminum metal-matrix composites such as Metamic™ and aluminum-boron carbide laminate composites such as Boral®. The performance of neutron poison plates can be affected by aging mechanisms induced by various physicochemical, thermal-mechanical, and irradiation conditions, including different types of corrosion and blistering, creep, thermal aging, radiation embrittlement, and boron depletion.

Boron depletion as aging mechanism refers to the reduction in the neutron absorption capacity of a material due to the loss of boron when exposed to neutron fluence. In neutron poison materials such as borated stainless steel, borated aluminum alloys, and aluminum-based composites, ^{10}B capture neutrons, becoming excited ^{11}B , which then decay into high-energy alpha particles and ^7Li . In this reaction, one neutron depletes one ^{10}B . According to Managing Aging Process in Storage report, at typical neutron flux and ^{10}B concentration levels, neutron exposure over 60 years could deplete up to 0.0002% of the available ^{10}B . Even under the most conservative scenario, using the highest expected neutron flux and the lowest ^{10}B concentration, only 0.02% of the available ^{10}B would be depleted after 60 years, which is insufficient to significantly decrease the criticality control function of the neutron absorber.

In this study, it was determined that the impact of boron depletion on criticality control functions in U.S. commercial DSSs is minimal. A comprehensive investigation and analysis were conducted to evaluate methods for assessing boron depletion across various DSSs. Furthermore, using the KENO V.a module of the SCALE code, the temporal changes in boron content and the corresponding effective multiplication factor (k_{eff}) of Metamic™, a neutron absorber used in actual DSSs, were calculated.

The study identified the impact of variations in elemental composition and ^{10}B areal density on the k_{eff} of commercial Metamic™. Furthermore, by assuming various elemental compositions, the relationship between ^{10}B areal density and k_{eff} was compared, confirming that a decrease in ^{10}B areal density leads to an increase in k_{eff} . These findings offer valuable insights into the potential effects of boron depletion on criticality safety.

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Keywords: Dry storage system, Neutron absorber, Boron depletion, Effective multiplication factor, Criticality

Preliminary Radiation Shielding Evaluation for Conceptual Design of PCRf

Siwan Noh*, Woo Jin Jo, and Jonghui Han
Korea Atomic Energy Research Institute (KAERI)
*snoh@kaeri.re.kr

Pyroprocess has been studied as an option technology for spent fuel management, and it is necessary to ensure the practicality of the developed technology. To this end, conceptual designs are being carried out for the preparation of a Pyroprocess Commercializing Research Facility (PCRf). PCRf handling spent nuclear fuel inevitably generate large amounts of photons and neutrons, so proper shielding of the hot cells is essential. As the shielding wall thickness of the hot cell is required as an input for establishing the general arrangement and calculating the facility area in the conceptual design stage, this study calculated the shielding wall thickness for PCRf hot cells. The hot cells subject to shielding analysis are headend cell, electrorefining process cell, U/TRU storage cell, waste treatment cell, and waste storage cell, and not only the side shielding wall but also the upper ceiling thickness was calculated. First, cell and device-specific inventories were calculated and used to obtain photon and neutron energy spectra. Using the obtained spectra and the MCNP computer code, the dose rate per shielding wall depth was calculated and compared to the dose rate target for each zone to determine the required shielding wall thickness. Considering the availability and constructability of concrete, ordinary concrete rather than heavy concrete was applied. The shielding wall thickness data for each hot cell calculated in this study will be used as an input for the conceptual design of the PCRf.

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Keywords: Pyroprocess, PCRf, Spent fuel, Radiation shielding, Hot-cell, MCNP

Review of Fuel Handling Events in Nuclear Power Plants: System Reliability, Event Management, and Operational Guidelines

Sanggyoon Chang¹, Junkyu Lee¹, Mingyu Kim¹, Jin Yoo², Changsok Cho², and Daehyuk Kwon³

¹KEPCO Engineering & Construction Company (KEPCO E&C)

²Nuclear Eng.

³Korea Hydro & Nuclear Power (KHNP)

*sgchang@kepc-eenc.com

This study reviews fuel handling events at nuclear power plants. The paper provides technical information on events that occur during the process of handling nuclear fuel, along with the corresponding operational manuals.

The fuel handling system comprises components and equipment designed to handle nuclear fuel. The primary function of this equipment is to lift (or load) fuel assemblies from (or into) the reactor and transfer them to the spent fuel storage pool during refueling outages. The fuel handling system is engineered to prevent criticality accidents, radiological release from mechanical and thermal damage to irradiated fuel, and unacceptable radiation exposure to personnel, in accordance with design requirements, regulatory guidelines, and engineering codes and standards. To meet these requirements, the equipment must be designed to handle fuel assemblies with a high degree of reliability. This reliability and efficiency can be achieved through simple and direct operation. All machines in the fuel handling system are networked together to facilitate the communication of machine status and other pertinent information between them.

The fuel handling events discussed in this study are summarized as follows: 1) BTI (Bridge and Trolley Interlocks) algorithm error during off-index operation at the reactor, 2) Malfunction of interlock control between the refueling machine and the fuel transfer system upender, 3) Damage to nuclear fuel during lifting or rotating it on the spent fuel storage rack, 4) A postulated accident involving nuclear fuel stuck in the fuel transfer tube and hoist box of the refueling machine, and 5) Drindown in the refueling cavity due to a leak in the pool seal assembly while filling the cavity with cooling water.

During a refueling outage, handling nuclear fuel between the reactor and the spent fuel storage rack is a critical process, necessitating preparedness for potential events. The fuel handling equipment is equipped with PC-based PLCs, and the database provides operational information for these events along with self-diagnostic functions. This operational information, including event logs stored in the database, serves as a valuable guideline for operators to take appropriate actions according to the corresponding manuals in response to these events.

Keywords: Nuclear fuel handling events, Fuel handling equipment, Spent fuel handling machine, Refueling machine, Refueling outage

Study on Radiological Impact of UT Operation in Spent Nuclear Fuel Pool Environment

Hakyun Lee*, Changsun Kim, Wonil Kim, Namyong Park, and Jeyong Choi

enesG Co., Ltd.

*Hakyun.lee@enesg.co.kr

Ultrasonic Testing (UT), applied to the detection of defects in spent nuclear fuel, is used for distinguishing between minor defects in first-cycle, second-cycle, and discharged fuel on the storage racks. The basic principle of UT involves transmitting and receiving ultrasonic pulses using a pair of ultrasonic transducers, which detect the presence of water in the fuel cladding caused by defects, thereby determining the existence of defects.

Since UT is conducted in close proximity to spent nuclear fuel assemblies due to its operational characteristics, there are direct radiological impacts from gamma rays and neutrons emitted from the spent nuclear fuel. These effects are particularly concentrated on the ultrasonic transducers entering between the plate supporting the assemblies and the fuel rods, highlighting the need for studies on equipment lifespan evaluation and radiation protection for the operators.

In this study, we aim to use PHITS (Particle and Heavy Ion Transport code System) to evaluate the lifespan of UT equipment and assess the extent of radiological impact. Initially, we designed the geometry of specific UT equipment through PHITS, and we plan to use ORIGEN-ARP to calculate gamma-ray source terms for the spent nuclear fuel based on its burn-up cycle.

Although this research is still in progress, once the data is obtained, it could be actively utilized in the radiological evaluation of UT equipment used for inspecting the 177 fuel assemblies in OPR (Optimized Power Reactor). Furthermore, it may be incorporated into work procedures to enhance radiation protection for operators handling the equipment.

Keywords: Ultrasonic Testing (UT), Radiological impact, Worker, Spent fuel

Review on Probability Risk Assessment of Dry Cask Storage System for Spent Nuclear Fuel in the United States

Dong-Jin Kim* and Sukhoon Kim

FNC Technology Co., Ltd.

*dj960930@fnctech.com

In accordance with the recalculated result for saturation forecast of spent nuclear fuel (SNF) based on the 10th basic plan for electricity demand and supply, it is anticipated that the on-site storage capacity for SNF will be saturated sequentially, starting with Hanbit nuclear power plant in 2030. In response to this issue, efforts are underway to secure interim storage facility for high-level radioactive waste in Korea. For the SNF storage facility, since compliance with safety objectives specified as exposure dose and/or risk shall be demonstrated, methodology for quantitative (i.e., deterministic and probabilistic) safety assessment should be developed. To date, few probabilistic safety assessments (PSAs) for the SNF interim storage facility have been conducted in Korea. Accordingly, in this study, we reviewed pilot application of PSA methodology to dry storage systems for SNF in the United States, focusing on case studies documented in EPRI-1002877 and NUREG-1864.

Based on review results, we identified steps of PSA and principal inputs for each step. The first step is to identify the accident scenarios. This involves deriving the frequency of initiating events (considering both potential accidents within the facility and external events) and their conditional probabilities. The second step is to estimate the release source term for the scenario. This is calculated by multiplying the nuclide inventory by the release fraction from SNF. The nuclide inventory is determined using computational code such as ORIGEN. The release fraction is calculated as the product of fuel damage rate, release rate of fuel rod to cask, and release rate of cask to environment. The third step is to conduct the consequence analysis applying the release source term. Consequences are calculated using codes such as MACCS2 and RISKIND, which rely on atmospheric dispersion analysis and are expressed as the risk of early fatalities and latent cancer fatalities among receptors located at specific distances. Key input data include release height, weather conditions, population distribution, and code relevant factors. The final step is to assess the risk for the dry storage facility. The risk is derived by multiplying the accident probabilities and the consequence. To ensure the comprehensive safety, the risk assessment must account for all potential impacts throughout loading, transfer, and on-site storage phase.

This study could provide a foundation for developing PSA models and technical infrastructure for SNF interim storage facility in Korea. However, it is required to derive input data reflecting domestic circumstances for the analysis.

Keywords: Probability risk assessment, Consequence analysis, Dry storage system, SNF interim storage

The Impact Analysis of Concrete Structure in Dry Storage Facilities After Fire Accident

Seungpil Kim and Sang Soon Cho*

Korea Atomic Energy Research Institute (KAERI)

*sscho96@kaeri.re.kr

In the case of dry storage facilities, lots of dangerous phenomena are existed that are able to make widespread safety problem. For this reason, in dry storage facilities, evaluations of safety under accidental conditions are important. Accidental conditions in dry storage facilities include natural phenomena such as floods, tornadoes, tsunamis, typhoons, earthquakes, and artificial phenomena such as airplane crashes, fire. However, among phenomena, fire is one of phenomena that is focus on temperature. High temperature is affected on not only radioactive materials but also facilities materials like concrete. The concrete is degraded by high temperature and it lead to loss of facilities strength. So, degraded material properties should be concerned.

In this paper, thermal analysis was performed based on concrete materials. The reasons of degradation by temperature in concrete were confirmed using physical mechanism and chemical mechanism. Degraded compression stress, degraded elastic modulus, degraded compression strain and degraded tensile stress (split test) were confirmed and formula were suggested to predict degradation material properties at target temperatures. Furthermore, thermal analysis were performed to predict concrete temperature under 800°C temperature, 30 minutes and 3 hours relaxation conditions. Concrete wall model was used in thermal analysis and temperature distribution was evaluated after thermal analysis. Furthermore, impact analysis was performed to evaluate the safety of concrete wall which was degraded by fire condition.

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Keywords: Concrete storage structure, Fire, Concrete, Degradation, Missile impact analysis

Functional Recoating on Aluminosilicate Foam Filters for Gaseous Cesium Capture

Byoungjin So*, Seok-Min Hong, and Jae Soo Ryu

Korea Atomic Energy Research Institute (KAERI)

*byoungjinso@kaeri.re.kr

The reprocessing process of spent fuel releases radioactive gases, with cesium-137 (^{137}Cs , $T_{1/2} = 30.2$ y) being of particular interest due to its long half-life and large quantity. To collect gaseous cesium, a reticulated ceramic foam filter with kaolin-based filter is used, capable of capturing cesium above 671°C (boiling point for metallic Cs). However, the foam filter has structural limitations such as broken struts and cracks due to the burnout of the sacrificial template during the synthesis process.

Here, additional functionalization through the addition of TiO_2 or MoO_3 to ceramic slurries for recoating was investigated. The functional recoating on the struts was performed using centrifugation and post-sintering to cover the cracks and showed different crystallization behaviors with each dopant, demonstrating improved mechanical and capturing performances compared to undoped specimens. The microstructure, mechanical strength, and cesium-capturing performance after the functional recoating were thoroughly investigated.

Keywords: Cesium, Functionalization, Off-gas filter, Reticulated ceramic foam filter, Coating

Input Material Accounting With Simplified Sampling-Homogenization Procedure for Spent Nuclear Fuel Assembly

Seonkwang Yoon, Seong-Kyu Ahn, Byung-Hee Won, and Chaehun Lee*

Korea Atomic Energy Research Institute (KAERI)

*sy@kaeri.re.kr

It is essential to secure the safeguardability of pyro-processing treating special nuclear materials (SNMs), e.g., Pu and U, in terms of nuclear non-proliferation. Because spent nuclear fuels (SNFs) contains significant amounts of SNMs, evaluating the material balance at the head-end process, so called input material accounting is necessary. Non-destructive assay (NDA) could be theoretically advantageous when bulk target samples are measured. However, the entire spent fuel assemblies are too complicated to be simply examined by NDA. One of the reasons is that the detection responses of an NDA instrument can be varied dependently on the geometric properties of a target of to be measured, such as nonuniform density and axial/radial distributions of SNMs. This could lead to difficulty to achieve sufficient measurement precision. Therefore, the total mass estimation of SNMs contained in such SNF can be carried out by destructive assay (DA) by sampling several small samples.

To expand the investigated results for small DA samples (several grams) to the entire SNF assembly (≈ 500 kg), the representativeness of the samples should be considerably concerned. Therefore, an appropriate procedure has to be preceded including sampling and hominization processes. The main purpose of this procedure is to reduce the sampling uncertainty, that is the variance of SNM concentrations confirmed among different DA samples. A previous study employed the two-stage homogenization method consisted of successive sampling-homogenization process as twice, and it achieved sufficiently low sampling uncertainty. Nevertheless, the homogenization process for the entire SNF assembly could lead to time/cost/task-burden issues.

In this study, therefore, a simplified procedure was suggested by employing single sampling-homogenization process. It was confirmed whether sampling masses of several-kilograms from the entire SNF is enough to achieve 1% of Pu sampling uncertainty for the suggested procedure. To this end, diverse parameters which can affect on Pu sampling certainty were considered, such as the efficiency of mechanical decladding, sampling mass, and particle size.

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Keywords: Pyro-processing, Safeguards, Nuclear material accounting, Sampling uncertainty, Homogenization, Heterogeneity

Repackaging Concept for PWR Spent Nuclear Fuel Prior to Disposal

Sang-Hwan Lee*, Chang-Min Shin, Hyungoo Kang, and Tae-Chul Moon

Korea Radioactive Waste Agency (KORAD)

*mlsh87m@korad.or.kr

Spent nuclear fuel (SNF), which has undergone nuclear reaction in a reactor, is a high-level radioactive waste that emits significant decay heat and radiation. Most countries plan to retrieve SNF stored interim storage facilities and repackaging it into disposal cask at surface disposal facilities. Repackaging process involves emplacing the SNF into cask designed to meet the technical requirements of each management stage. In Finland and Sweden, leading countries of repackaging technology for SNF disposal, they have been received construction permissions for disposal facilities, are currently building repackaging facilities. Meanwhile, countries like the United States and Canada have conducted conceptual designs for such facilities. In contrast, South Korea has done limited studies on repackaging likely due to uncertainties about storage cask, facility timelines and the amount of SNF. This study aims to establish the conceptual layout and operational procedures for a basic repackaging facility, taking into account national policy directions and ongoing SNF management projects at nuclear power plant sites. Additionally, it aims to develop detailed repackaging processes to estimate the annual disposal capacity based on the amount of SNF generated.

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Keywords: Repackaging, Spent nuclear fuel, Conceptual design, Storage, Disposal

Study on Calculation Methodology for Burnup Using Characteristics of Spent Fuel

SeongHee Park*, SeungUk Yoo, and MoonOh Kim

KONES Corp.

*shpark@kones21.com

Burnup of spent nuclear fuel serves as a primary indicator for managing spent nuclear fuel. However, accurately measuring burnup is challenging due to the risks associated with handling spent nuclear fuel and the influence of various factors. This study aims to develop a nondestructive method for measuring burnup by deriving a calculation approach based on the characteristics of spent nuclear fuel. Preliminary research involved performing depletion calculations using ORIGEN to establish a database that considers various nuclear fuel properties.

In this study, the database was used to analyze trends among various data points and develop a regression model for determining the relationship between the characteristics of spent nuclear fuel and burnup.

The factors that can represent the characteristics of spent nuclear fuel include the ratios of specific isotopes, cooling time, and initial enrichment. Due to the complex relationships between these factors and burnup, the resulting regression model was formulated as a nonlinear function.

A regression analysis code incorporating the Gauss-Newton method for nonlinear functions was developed. This code enables the derivation of a burnup calculation function using only a limited set of calculation data.

The derived burnup calculation function was rigorously validated against test set, achieving an error margin of less than 5%. This demonstrates the model's robustness and precision in predicting burnup based on key isotopic and operational parameters. The successful application of this advanced nonlinear regression methodology suggests that the model can serve as a foundational tool for quickly and accurately calculating the burnup of actual spent nuclear fuel, potentially aiding in the management and safety assessment of nuclear fuel cycles.

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Keywords: Burn-up, Spent fuel, Inventory, Characteristics

Probability Distributions of Crack Depth and Length for Safety Analysis in Spent Nuclear Cladding

Sangil Choi*, Seunghwan Yu, and Sangsoon Cho

Korea Atomic Energy Research Institute (KAERI)

*sichoi89@kaeri.re.kr

In order to evaluate the safety of spent nuclear fuel after vacuum drying, it is very important to establish the database on the crack size distribution. In the previous study, only the probability distribution of crack size according to depth was calculated by considering pellet cladding interaction. In this study, we attempted to derive the probability distribution of cracks considering not only the depth of cracks but also the length.

The crack size distribution is a necessary factor for safety evaluation not only in cladding but also in all pipe-shaped structures. Since a study on the probability distribution of cracks including length has already been conducted for the probabilistic safety evaluation of LOCA accidents (NUREG-2189 report) in the primary system of a nuclear reactor, we attempted to apply the logic of LOCA study to cladding.

In the NUREG-2189 report, the probability distribution according to depth was first derived. Then, the probability distribution of length according to depth was presented. Therefore, a function dependent on crack depth and length could be derived by simply multiplying the probability of crack depth and the probability of crack length. In future studies, we will try to conduct a probabilistic safety evaluation through FEM simulation that reflects both the depth and length of cracks.

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Keywords: Spent nuclear fuel, Cladding integrity, Probabilistic safety assessment, Crack, Crack distribution

Crushing and Mixing Processes of Uranium Powder to Make Sound Green Pellets

Yung Zun Cho*, Jae Won Lee, Kweon Ho Kang, and Joo Ho Lee

Korea Atomic Energy Research Institute (KAERI)

*choyj@kaeri.re.kr

The term “damaged spent fuel” refers to spent nuclear fuel that requires non-standard handling to meet relevant safety, regulatory and operational requirements. Canning is a commercial technology used for the transportation and both short-term and long-term storage of spent fuel. However, while canning technology has been shown to be safe for transport and storage, its reliability for final disposal has yet to be demonstrated. In particular, it is essential to develop stabilization technologies that ensure the safety not only of transport and storage, but also of permanent disposal for severely damaged fuel, rod cuts, pellets leaking from fuel rods, fragment and powder. The Korea Atomic Energy Research Institute (KAERI) is developing technologies to ensure the stability of severely damaged nuclear fuel. This stabilization technology involves several steps: recovering the fuel from the damaged rod, processing it into new pellets, placing these pellets into new rod, and welding. Among these processes, converting of uranium powder recovered from the oxidative decladding process into stable pellets is the most critical step. In this study, characteristic analysis experiments were carried out on the crushing and mixing processes that must be carried out prior to pellet pressing to produce sound green pellets. In particular, the possibility of simplifying the process was explored by performing the crushing and mixing processes within the oxidative decladding device using an air cylinder-based vibration mechanism, thus eliminating the need for crushing or mixing equipment.

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Keywords: Damaged spent fuel, Stabilization, Uranium powder, Crushing, Mixing

Flow Regime and Off-gas Capturing Characteristics of Molten Hydroxide Salt Scrubber

Yung Zun Cho*, Do Kyu Kang, Seok Min Hong, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*choyj@kaeri.re.kr

The off-gas of MSR is expected to contain noble and reactive gases, aerosols, particles, and volatile species. As one of the continuous parallel off-gas capturing equipment, a molten salt scrubber has been considered for removing (or capturing) some species from the MSR off-gas stream. A molten hydroxide (NaOH-KOH eutectic) scrubber can trap (or neutralize) particles (noble metals, aerosols of fuel salt, etc.) and reactive gases (HCl, Cl₂, I₂, HI, etc.). Considering operational simplicity and long-term safe operation, a liquid-batch bubble column (off-gas and molten hydroxide two-phase flow system) was selected and is under development. In a two-phase flow system, the flow regime characteristic is essential from various perspectives: heat and mass transfer, axial and radial mixing. In this study, the effects of He gas flow rate on flow regime transition have been studied by the pressure drop method to find out proper operating conditions in the NaOH-KOH molten salt scrubber. Additionally, the basic capturing characteristics of reactive gases (HCl, I₂) were evaluated.

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Keywords: MSR, Off-gas capturing, Molten hydroxide scrubber, Flow regime transition, Capturing characteristics

Simulation of Bending Behavior of High Burnup Spent Nuclear Fuel Rod Considering Progressive Breakage of Interfacial Bond

Guhyeon Jeong and Sanghoon Lee*

Keimyung University

*shlee1222@kmu.ac.kr

Maintaining the integrity of spent nuclear fuel (SNF) during storage and transportation is critical for its safe and economical management. Due to the high radioactivity and decay heat of spent nuclear fuel, computational analysis is often used to assess the safety and risk of SNF-related activities. Assessing the mechanical integrity of SNF rods in various conditions is essential for both risk and fuel retrievability evaluation. However, the altered characteristics of SNF rods due to irradiation during reactor operation pose significant challenges, as obtaining this data is technically and economically difficult. Experimental research on irradiated fuel cladding has been conducted in hot cells to address this gap. The Oak Ridge National Laboratory (ORNL) of the U.S. Department of Energy developed the Cyclic Integrated Reversible-Bending Fatigue Tester (CIRFT) to characterize the bending behavior of high-burnup spent fuel rods and quantify their fatigue limits. Their findings revealed that the interfacial bonding within SNF rods plays a critical role in their resistance to external forces and that this bond progressively degrades under load, altering the rods' flexural rigidity. Previous modeling efforts typically assumed the interface to be either fully bonded or fully debonded, which does not reflect real-world conditions. In this study, we developed a finite element model of high-burnup SNF rods that simulates the progressive degradation of interfacial bonding. Using cohesive zone modeling in ABAQUS, we modeled the bonded interface and calibrated the parameters through optimization to minimize discrepancies between the calculated and CIRFT-obtained moment-curvature curves. This research provides a more realistic prediction of SNF rod behavior and enables more accurate damage ratio calculations under various conditions, including shock. Additionally, the proposed inverse engineering approach offers insights into the unknown properties of pellet-cladding and pellet-pellet bonds.

Keywords: Spent nuclear fuel, Finite element analysis, Interface damage, Cohesive elements, Safety assessments

Classification of Interim Dry Storage Methods of PWR Spent Nuclear Fuel in Korea

Oujeong Yoo*, Gyuyeon Kim, and Kiyeoul Seong

Korea Radioactive Waste Agency (KORAD)

*dnwjddl00@korad.or.kr

According to the 2023 spent nuclear fuel (SNF) saturation forecast results based on Korea's 10th Basic Electricity Supply and Demand Plan, on-site SNF storage facilities are expected to be saturated sequentially, starting with Hanbit NPP in 2030. As part of efforts to resolve this issue, siting process for a management site for high-level radioactive waste is under preparation in Korea, including interim storage facilities for SNF.

Wet and dry storage facilities are broadly used in many countries, and a wide variety of dry storage systems are currently utilized. In Korea, dry storage systems are considered viable because of their safety, as proven by operational experience globally. It is necessary to comprehensively consider various interim storage methods applicable to Korea, considering the long-term uncertainties of interim storage projects. However, since it is only possible to consider some options within the limited time available in the preliminary planning stage, this study reviewed the current status of the interim storage facilities in each country. Based on the review, the interim storage systems were grouped. However, dry storage for heavy water reactor SNF was excluded from this analysis because the module MACSTOR/KN-400 is commercially available and operational, and it has fewer options than PWR SNF storage.

The following factors were considered in this grouping process, taking into account the receipt process for interim storage of SNF in general: (a) type of transport packaging (cask for transport only, dual purpose cask for transport and storage (DPC), or canister); (b) characteristics of on-site storage (spent fuel packaging process, whether requiring transfer cask, and subjects for preliminary inspection); and (c) characteristics of interim storage (subjects of initial receipt, whether requiring discharging of SNF assemblies or canister, subject to inspection, etc.).

As a result, the storage methods of PWR SNF were classified into three groups, and the characteristics of each group are summarized as follows: (a) Group 1: Spent fuel is sealed in a DPC and stored in the interim storage facility as it is after receipt; (b) Group 2: Transport package, which contains a sealed canister where SNF assemblies are loaded, is received, then the canister is unloaded and placed in an interim storage facility; and (c) Group 3: Transport package, where SNF assemblies are loaded, is received, and SNF assemblies are unloaded, then either in sealed canisters or encapsulated in separate containers, and placed in an interim storage facility.

Keywords: Interim dry storage, PWR spent nuclear fuel, Storage methods, Type of transport packaging, Characteristics of on-site storage, Characteristics of interim storage

Design and Fabrication of a Heater Module With a Readily Detachable/Attachable Structure

Byung Suk Park*, Jong Kwang Lee, Seung Nam Yu, and Jong Hui Han

Korea Atomic Energy Research Institute (KAERI)

*nbspark@kaeri.re.kr

This paper introduces a mechanism for easily detaching and attaching heaters in inaccessible environments using cranes or gantry robots. The heater system consists of a two-part heater module, a heater module drive unit, and a carriage. The heater module comprises two hexagonal heaters arranged symmetrically, facing each other. The heater module drive unit simultaneously moves each heater in opposite directions (moving away from each other) and towards each other. The drive unit is installed on the carriage, and the heater module is positioned on top of the drive unit. Each heater module top is fitted with an I-bolt shaped to allow for hooking by a crane, positioned at the center of gravity. Additionally, two-headed rods are attached to enable lifting by a gantry robot gripper. Each module's heater is composed of two zones with distinct temperature ranges. The heaters are powered by a 3-phase 440 V supply, with an upper heater output of 13 kW and a lower heater output of 6 kW. The heater is designed and manufactured to operate at a maximum temperature of 1,000°C, with an external surface temperature kept below 150°C, and an operational temperature of 900°C. The heater module upper and lower separation/connection interface consists of the heater module lower plate (upper interface) and the heater module drive unit upper plate (lower interface). A male electrical connector corresponding to the female electrical connector on the upper interface plate is attached to the plate of the lower interface. Two guide pins are attached to the plate of the lower interface so that they can be inserted into the holes of the upper interface plate. This mechanism is expected to make a significant contribution to the development of devices requiring maintenance in unmanned automated facilities.

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Keywords: Heater module, Detachable, Attachable, Crane, Gantry robot, Electrical connector

Multi-Physics Modelling of Radiation Induced Material Degradation in Dry Storages of Used Nuclear Fuel

Sangjin Kim, Samuel Park, and Sungyeol Choi*

Seoul National University (SNU)

*choisys7@snu.ac.kr

In South Korea, the capacity for wet storage of spent nuclear fuel is expected to reach its limit around the year 2030. This situation necessitates the adoption of dry storage solutions, utilizing specially-designed casks for more efficient and safe management of spent nuclear fuel. This study aims to analyze the thermal characteristics and performance of the KORAD 21 cask, developed by the Korea Radioactive Waste Agency (KORAD), as part of the shift towards dry storage solutions. A key focus of the study is the investigation of the thermal characteristics that influence the cask's longevity and safety, particularly concerning the potential for corrosion induced by radiation: which is one of the major challenges in dry storage. The study models heat conduction, convection, and radiation using the multi-physics software, COMSOL. Convection is modeled through the heat transfer coefficient, which accounts for the forced convection on the cask's external surface.

The temperature distribution within the cask was modeled over 50 years, using a heat flux equation to represent the decay heat from the spent nuclear fuel. The nuclear fuel considered in this study has a burn-up of 45,000 MWd/t and an enrichment of 4.0wt%.

The simulation results indicated that cask reached maximum temperature of approximately 594 K after 150 hours. Over the course of 50 years, this maximum temperature gradually decreased to around 400 K, reflecting the natural decay of the fuels heat output.

This thermal behavior suggests the potential for new design modifications that could improve heat dissipation. One proposed modification includes reducing the number of baskets within the cask, thereby lowering the total uranium mass and enhancing cooling efficiency. Another option is to explore the use of materials with higher thermal conductivity to better dissipate heat. Additionally, replacing helium with nitrogen as the filling gas in the cask could be beneficial, as Nusselt number of nitrogen is approximately four times higher than helium, indicating more efficient convective heat transfer.

These results aid in providing a better understanding of the thermal processes in dry storage casks, which is crucial for developing more effective strategies and ensuring long-term safety and visibility of dry storage systems enabling more effective prevention strategies.

Keywords: Canister, Simulation model, Numerical analysis, Dry storage

Pore Characteristics Effects on Technetium Capture in CaO-based Adsorbent

Namcheol Kim^{1,2}, Chang Hwa Lee², Eun-Young Choi^{1,2}, and Seok-Min Hong^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*seokminhong@kaeri.re.kr

The operation of nuclear power plants globally leads to the production of a significant amount of spent nuclear fuel. The handling methods for this spent fuel can be categorized into three main approaches: storage, direct disposal, and recycling. Pyroprocessing is a technology specifically designed to recycle accumulated spent nuclear fuel. During pyroprocessing, various fission products (FPs) such as C-14, H-3, I-129, and Cs-137 are generated. Among these, technetium (Tc-99) is particularly notable as it is a gaseous nuclear isotope with a long half-life and high mobility in the form of TcO_4^- in aqueous solutions. This makes it crucial to capture Tc-99 strictly to prevent radioactive contamination of the environment.

Previous studies have utilized ion-exchange or adsorption using Metal Organic Frameworks (MOFs) to remove Tc-99. However, these methods are designed for separation in aqueous solutions, not in the gaseous state. In this study, a CaO-based adsorbent was developed for capturing rhenium (Re) as a surrogate for radioactive Tc-99. Isopropyl alcohol (IPA) was employed as a pore-forming agent during the preparation of the adsorbent, and its effects on the characteristics and adsorption performance were investigated. The pore sizes were analyzed using nitrogen (N_2) adsorption isotherm analysis and mercury (Hg) intrusion curves. The results confirmed that the addition of IPA significantly influenced the formation of macropores. This macroporous structure was found to enhance the adsorption performance for Re.

Keywords: Spent nuclear fuel, Technetium, Fission product, CaO adsorbent

Review on Strategies of Sustainable Advanced Nuclear Fuel Cycles

HunSuk Im

Korea Atomic Energy Research Institute (KAERI)

ihs95@kaeri.re.kr

Advanced reactors can be categorized based on their use of various coolants, fuel types, and neutron spectra, with key examples including small modular light water reactors (LWRs), sodium- and lead-cooled fast reactors, and gas- and salt-cooled high-temperature reactors (FHRs). These reactors offer several advantages over traditional reactors, such as lower costs, faster construction, smaller size, inherent safety features, and reduced waste production. And research on their fuel cycles plays a critical role in establishing a low-carbon energy system to combat climate change.

However, several challenges must be addressed, including economic viability, safety, regulatory frameworks, and non-proliferation concerns. Notably, most advanced reactors rely on high-assay low-enriched uranium (HALEU) for fuel, and more than 50% of the global commercial HALEU supply currently depends on Russia, raising national security concerns. In addition, the deployment of advanced reactors must be balanced across various factors, such as economic feasibility, public acceptance, deployment scenarios, system integration and safety, and market readiness. The strategic implementation of advanced reactors requires collaboration between governments and private industries to ensure successful outcomes.

In Korea, efforts have been made to manage spent fuel from light water reactors through pyroprocessing and deep geological disposal research. Recently, research has expanded to explore various processing technologies and new disposal system combinations, alongside the development of evaluation programs based on selected criteria to determine optimal management strategies for spent fuel and high-level radioactive waste. These efforts are expected to serve as a foundation for future management strategies for spent nuclear fuel and high-level waste of advanced reactors.

Thus, a strategic approach to the back-end of the nuclear fuel cycle for advanced reactors is deemed crucial. This review will explore strategies related to advanced reactors and nuclear fuel cycles across different countries, as well as aim to establish the necessary strategies for the back-end nuclear fuel cycle of advanced nuclear reactors in the domestic context.

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Keywords: Nuclear fuel cycle, Advanced nuclear reactors, Back-end nuclear fuel cycle, Management, High-level radioactive wastes

Fabrication of Helium Circulation Automatic Control Experimental Device

Byungsuk Park*, Dokyu Kang, Seok-Min Hong, Yung Zun Cho, Joo Ho Lee, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*nbspark@kaeri.re.kr

In Molten Salt Reactors (MSRs), maintaining the high purity of molten salt is crucial. To achieve this, it is necessary to capture oxygen, moisture, and noble gases such as Xe and Kr. These are typically captured by circulating an inert gas, usually helium (He). This paper presents a device designed to capture only oxygen and moisture while circulating He. The purpose of this device is to produce fundamental experimental data for designing the He circulation loop in MSRs. The paper introduces the experimental setup and the following results: First, since the He circulation loop involves many mechanical connections, there is a potential for leaks. Therefore, we conducted a leak test to determine the acceptable leak rate. Second, the circulating He requires appropriate pressure, and we demonstrate that stable pressure can be maintained through automatic control. Third, the circulating He must maintain an optimal flow rate, and we propose the best flow control method. Fourth, when oxygen and moisture are introduced into the circulating He, we present the performance of the capture material and the optimal automatic control method. These experimental results are expected to significantly contribute to the design of the He circulation loop for maintaining high purity in future MSRs.

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Keywords: Molten salt reactor, He circulation, Leak rate, Automatic control, Flow rate

Designing Transport Cask for HANARO Research Reactor Spent Fuel

Kyung Min Kim, Min Sik Jeon, Soo Hyeok Lee, Hwi Joon Jeong, and Kyung Taek Lim*

Sejong University

*kl2548@sejong.ac.kr

While extensive research has been conducted in Korea to develop casks for transporting spent fuel from nuclear power plants, the same level of attention has not been given to designing casks for research reactors. This is a significant gap, considering that the nuclear fuel used in research reactors is considerably more enriched than that used in power reactors. Therefore, designing a cask to transport and store spent fuel from a research reactor presents unique challenges that require careful consideration. In particular, the task demands a design that can effectively handle high enrichment. Among the commercially available transport casks, the TN-LC transport cask made by ORANO stands out as a promising model that can accommodate spent fuel from the HANARO research reactor in Korea. In this study, we explored using different materials in the TN-LC cask to identify lighter materials that can enhance radiation attenuation properties while maintaining safety in terms of criticality.

Recent advancements have highlighted epoxy and clay-based materials as potential candidates for radiation shielding. In this study, we designed and evaluated TN-LC transport casks with a combination of epoxy, clay, and metal-based materials, comparing their safety with conventional TN-LC transport casks. The ORIGEN code was used to generate the source term for HANARO spent fuel, while the MCNP simulation code was utilized for criticality and shielding calculations. We also simulated Westinghouse 17×17 spent fuel to ensure the transport cask accommodates different fuel types. The F2 Tally was employed to measure the effective dose at the cask's surface, while the KCODE was utilized to determine the effective multiplication factor. These assessments were conducted to ensure the transport cask's safety when containing HANARO and Westinghouse spent fuel.

The results showed that casks made from lightweight materials like epoxy and clay were less effective than metal ones. Although these lightweight casks had a larger radius, they also had a higher surface dose rate compared to metal-based casks. These findings provide important insights into the choice of materials for developing transport casks in South Korea and establish a benchmark for assessing the risks of storing spent fuel from research reactors in transportation containers.

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Keywords: Spent fuel, Transport cask, MCNP, Gamma shielding, Neutron shielding

Evaluation of Radionuclide Inventories and Instant Release Fraction for ACE7 Type Fuel Assembly Using Serpent 2 Code

Kwang Pyo Choi and Ser Gi Hong*

Hanyang University

*sergihong@hanyang.ac.kr

In South Korea, Spent Nuclear Fuel (SNF) from PWR is currently stored in on-site wet storage facilities. However, as these wet storage facilities approach saturation, there has been a growing need to research disposal methods for SNF. Although South Korea has not yet finalized a plan for the ultimate disposal method, several countries, such as Sweden and Finland, are considering deep geological disposal (DGR). In line with this trend, the South Korean government is planning a long-term program to research essential technologies for DGR. Consequently, research and development on deep geological disposal of SNF has been ongoing, led by the Korea Radioactive Waste Agency (KORAD). The safety assessment of DGR concept has to analyze the whole processes during the canister breach leading the radionuclides release to the biosphere, which can be separated into the evaluating the source term, the simulation of transport process of radionuclides, the dose evaluation and so on.

The starting point of safety assessment for DGR is to evaluate the source term. This evaluation involves identifying which parts of the canister become the source term, determining which radionuclides should be the focus of the assessment, and estimating how much of these radionuclides might be released. Especially, there are two primary modes of radionuclide release. First, congruent release, is a very slow process where the nuclear fuel matrix dissolves into water due to the redox reactions with the substances like hydrogen peroxide formed by radiolysis. Second, instant release occurs when radionuclides, locally distributed within the spent nuclear fuel, are rapidly released due to severe damage to the canister.

In this work, we focus on evaluating the safety-relevant radionuclide inventory of low-burnup ACE7 type spent fuel, which is a domestic reference, based on previous research data. Furthermore, it will also estimate the IRF values for certain radionuclides, specifically ^{14}C , ^{59}Ni , ^{63}Ni , ^{93}Zr , ^{93}Mo , $^{93\text{m}}\text{Nb}$, and ^{94}Nb . These calculations were performed with the Monte Carlo code Serpent 2.

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Keywords: Spent Nuclear Fuel (SNF), Deep Geological Repository(DGR), Instant Release Fraction (IRF), Source term

Study on a Canister Lid Cutting Process for Repackaging of PWR Spent Nuclear Fuel

Chang-min Shin*, Hyun-goo Kang, Sang-hwan Lee, and Tae-chul Moon

Korea Radioactive Waste Agency (KORAD)

*auver97@korad.or.kr

Spent nuclear fuel (SNF) is transported to a disposal site after undergoing a long cooling period at an interim storage facility. The capacity of pressurized water reactor (PWR) SNF disposal containers currently being studied both domestically and internationally is significantly smaller than that of dry storage canisters. Therefore, to dispose of PWR SNF, repackaging into disposal containers at an encapsulation plant is necessary. Since the lids of SNF canisters are welded and sealed, a lid-cutting process must be performed in advance to extract the SNF within the hot cell of the encapsulation plant. As SNF may be exposed to air when the canister lid is cut, this process must be conducted remotely and automatically within a closed environment like a hot cell. Additionally, effective methods for transporting canisters with their lids cut and handling waste, such as the cut lids, must be considered. This study identifies the equipment required for efficient canister lid cutting of spent nuclear fuel and explores the processes involved in operating this equipment.

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Keywords: Spent nuclear Fuel (SF), Repackaging, Encapsulation plant, Cutting, Operational process

3분과

고준위폐기물처분 (ORAL)

Key Factors Determining the Capacity of a Multi-Purpose Canister in the Aspect of Geological Disposal of Spent Nuclear Fuel

Kwan Sik Chun

Korea Environ. Test and Anal. Lab.

kchun998@gmail.com

Since it will take at least 20 years for an interim storage facility of spent nuclear fuel to be opened in Korea, KHNP has started designing an on-site dry storage facility to store the fuels in this year. KORAD is developing a concept for a facility that would remove the sealed transportation-storage canister containing 32 PWR spent fuel assemblies from the transportation cask which this canister is inserted and repackage this canister. On the other hand, the roadmap of high-level radioactive waste management requires certain techniques of the on-site dry storage linked to the interim storage facility. These could be accomplished by using a dual-purpose canister or canister-inserted cask. Therefore, the domestic and overseas status of these kinds of canisters or casks for transportation, storage, and/or disposal has been surveyed, including loading capacity of spent nuclear fuel assemblies per canister or cask, and also single, dual, or triple purposes.

If the above concept would be linked to a repository, this canister should become a multi-purpose canister combined transportation-storage-disposal. Using this canister will be able to simplify handling, reduce radiation exposure of workers, lower the generation of secondary wastes, and also reduce the overall life-cycle cost. And so, although it is a little late because designing the on-site storage facility has been started, it is needed to conduct a detailed analysis on whether to use the multi-purpose canister as soon as possible before the on-site storage facility is built.

First of all, the capacity of the multi-purpose canister should be determined. This capacity will depend on the environmental conditions of a repository, for example, host rock, emplacement method (vertical, horizontal, or deep bore-hole), buffer materials and so on. Here, key factors, including thermal load, for determining the optimal capacity have been proposed. Also, some prerequisites and scenarios that should be considered when determining the maximum number of spent nuclear fuel assemblies that could be loaded into this canister have been suggested.

Keywords: Spent nuclear fuel, Multi-purpose, Thermal load, Canister, Storage, Disposal, Capacity

A Study on Hydrogeological Characteristics of Deep Fractured Rock in Korea

Hangbok Lee*, Chan Park, Dae-Sung Cheon, and Eui-Seob Park

Korea Institute of Geoscience and Mineral Resources (KIGAM)

*leehb@kigam.re.kr

For the successful site selection of deep geological disposal of high-level radioactive waste, a phased approach is crucial, along with the systematic selection and evaluation of geohydrological assessment factors suitable for the domestic geological environment. In this study, the hydraulic conductivity characteristics, which are considered the most critical evaluation factor in the field of hydrogeology, were assessed for deep rock aquifers where the disposal facility is expected to be located. Specifically, hydraulic tests were directly conducted in boreholes at depths of up to 500 meters and 750 meters in various rock types (granite, volcanic rock, gneiss, mudstone) distributed in Korea. Through this, in-situ pressure-flow data were obtained, and hydraulic conductivity values for each rock type and depth were derived using validated analytical methods. This comprehensive evaluation and analysis process of deep hydraulic characteristics represents the first of its kind in Korea. For this purpose, precision hydraulic testing equipment developed in-house was utilized, and detailed investigation procedures based on standard testing methods were applied to field tests. The analysis results showed that the average hydraulic conductivity values in granite, volcanic rock, and gneiss areas were in the range of 10^{-9} m/s, while in the mudstone area, the values were approximately 100 times lower (2 orders of magnitude) at an average of 10^{-11} m/s. Additionally, in the fractured rock aquifers of granite and volcanic rock areas, a slight decrease in permeability with depth was observed overall. In the gneiss area, the permeability differences were more significant depending on the rock type and the development of fracture zones rather than depth. In the mudstone area, where fractures were less developed, no significant variation or trend in rock permeability with depth was observed. The hydraulic conductivity data by rock type and depth presented in this study are expected to be utilized in building a foundational database for the site selection, design, and construction of disposal facilities in Korea.

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Keywords: Deep geological disposal, Deep rock aquifers, Hydrogeology, Hydraulic conductivity data by rock types and depth

Site Characterization of Small-scaled Field Infiltration Test Facility Near the KURT, KAERI

Won-Tak Joun*, Byeong-Hak Park, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*wtjoun@kaeri.re.kr

As the geological environment undergoes long-term evolution, changing surface environments caused by climate changes may progressively affect the stability of spent nuclear fuel disposal site. This influence arises primarily from the development of fluid flow pathways that establish connections between surface water system and deep geological formation, including deep aquifer. Understanding and managing these fluid flow dynamics across the entire geological system is crucial for minimizing uncertainties and ensuring the long-term safety of deep geological repository for nuclear waste. To address these concerns, it is imperative to verify the mechanisms governing fluid flow from the surface to the disposal site under similar geological conditions. By conducting rigorous testing and acquiring comprehensive data, researchers can enhance predictive models, improve repository design, and ensure the robustness of containment strategies over geological timescales.

POSIVA in Finland, a leading country in the geological disposal of spent nuclear fuel, has conducted the infiltration test to gather sufficient evidence supporting the long-term stability of the ONKALO repository. Similarly, research on the small-scaled field infiltration test is necessary to proceed with the construction of a repository and to secure its stability, and related research is being performed at the UNIT (UNsaturated zone In-situ Test) facility, KAERI (Korea Atomic Energy Research Institute).

The present study suggests the design and method of small-scaled field infiltration test based on the data on saturated/unsaturated zone characteristics. Although the results of those tests enabled us to obtain a quantitative representative value due to the specific site characteristics, it could help to understand some qualitative of a water penetrating mechanism and a hydrogeologic parameter in those kinds of sites where it is artificially made a reclaimed and flat land. Additional tests, including single-well push-pull and another type of infiltration test, are necessary near future to identify potentially uncertain hydrogeologic factors, and it is expected that small-scaled field infiltration test is considered to enrich the information available for the next stage of research about the long-term infiltration test and the simulation of a developed integrated hydrologic model.

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Keywords: Infiltration test, Tracer test, Pumping and recovery test, Hydrogeologic parameters

A Study on Geological Characteristics Using Deep Borehole Drilling by Tectonic Regime and Rock Types for Site Study for HLW Disposal

Dae-Sung Cheon*, Jai-Yong Park, You Hong Kihm, Hangbok Lee, and Seong Kon Lee

Korea Institute of Geoscience and Mineral Resources (KIGAM)

*cds@kigam.re.kr

In Korea, a variety of rock types, including granite, gneiss, shale, and volcanic rock, are distributed, necessitating thorough studies to determine the most suitable disposal rock type for HLW. Previous geological environmental surveys have predominantly focused on granite and surface investigations, with limited research conducted on deeper geological characteristics. To address this gap, there is a need for an in-depth study through deep borehole drilling investigations of granite, gneiss, shale, and volcanic rock types. To this end, KIGAM has secured two deep boreholes each year for five years starting in 2020 to investigate the deep geological characteristics of these major rock types. This study aims to briefly introduce the research conducted over the past five years. Among the evaluation parameters proposed in previous step-by-step site selection procedures, we identified key parameters crucial for determining the disposal rock type. We examined these key evaluation parameters in relation to tectonic regime, rock types, and both shallow subsurface and deep characteristics. Additionally, we reviewed the drilling survey, testing, and analysis technologies used in this study and developed a procedure for investigating and analyzing deep geological characteristics through deep drilling surveys. As a result of this study, a total of 10 deep boreholes were drilled, and data on key evaluation parameters were collected through field and laboratory tests. This data is available through the web service D-GIVES. Although the results from these 10 boreholes cannot be considered fully representative of all rock types in Korea, we believe they provide essential basic data that will be valuable for future site selection processes.

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Keywords: Geological characteristics, Deep geological disposal, Deep borehole drilling, Rock types, Tectonic regime

Analysis of Drilling Characteristics and Improvement of Borehole Deviation by Rock Types Based on Deep Drilling Experiences

Ki Seog Kim^{1,*}, Dae-Sung Cheon², and Jai-Yong Park²

¹HeesongnGeotek

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

*kskim@hsgeo.co.kr

High-level radioactive waste must be disposed at depths of less than 300 meters, and in South Korea, the expected disposal depth is approximately 500 meters. Therefore, understanding the characteristics of deep rock formations through drilling is essential for site selection and site characterization. As of now, South Korea has not yet determined the disposal concept or the host rock, and the geological environment is complex. Considering this, the Korea Institute of Geoscience and Mineral Resources has been conducting deep drilling at 750 meters depth on representative rock types in Korea, starting from 2020. The drilling sites were selected based on geotectonic regimes and rock types, with final deep drilling conducted on 3 granite holes, 3 sedimentary rock holes, 2 metamorphic rock holes, and 2 volcanic rock holes.

This study presents an analysis of drilling characteristics and borehole deviation based on the experiences gained from deep drilling. To understand drilling characteristics, various data acquired in real-time during drilling, such as drilling speed, torque, and inflow of water, were analyzed. Additionally, various methods employed to maintain the borehole deviation of the drilling holes, which are fundamental to drilling, will be introduced. In deep drilling, the final borehole position can vary significantly due to changes in deviation, making its management crucial. Particularly, borehole deviation can be influenced by rock type characteristics and drilling speed.

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Keywords: Drilling characteristics, Borehole deviation, Deep drilling, Rock types, High-level radioactive waste

Understanding Fault Zone Distributions Using Borehole Data: Insights From the KURT Study

Ji-Min Choi^{1,2,*}, Soolim Jung¹, Nak Kyu Kim¹, Kyung-Woo Park¹, and Young-Seog Kim²

¹Korea Atomic Energy Research Institute (KAERI)

²Pukyong National University

*jiminchoi@kaeri.re.kr

The deep host rock, an integral component of the natural barrier system, must remain isolated for prolonged periods to prevent the migration of radionuclides from high-level radioactive waste (HLW) disposal site into human environments. However, the secondary development of brittle structures, such as fracture zones and fault zones, after rock formation, creates primary pathways for groundwater-mediated nuclide transport. Therefore, understanding these brittle structures is crucial for evaluating the performance of natural barriers. Fault zones are areas of concentrated brittle deformation, and are typically surrounded by a high density of fractures. This study investigates the subsurface distribution of these structures by analyzing borehole data. The boreholes used in this study were drilled to a depth of approximately 200 meters and are arranged in a roughly square configuration with a horizontal spacing of roughly 100 meters around the KAERI Underground Research Tunnel (KURT) at the Korea Atomic Energy Research Institute (KAERI). To delineate the fracture and fault zones, we utilized both the core samples and borehole wall images obtained through the Borehole Image Processing System (BIPS). From the comparison of core sample data and fracture images from BIPS, we present a correlation between fracture density and fault distribution. This represents a preliminary step toward developing a comprehensive model of deep brittle structures. The findings from this study can be synthesized with fracture density analyses from previously drilled, deeper boreholes (depths ranging from 500 to 1,000 meters) to derive a robust, scaled-up model of deep brittle structures.

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Keywords: Fault zones, Borehole data, Brittle structures, Fracture analysis, KAERI underground research tunnel

Hydraulic Properties Measurement of Rocks for Deep Geological Disposal of High-Level Radioactive Waste

Dae Sung Lee^{1,*}, Seokhee Jung¹, Dae-Sung Cheon², and Junhyung Choi²

¹Dong-A University

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

*leeds@dau.ac.kr

Permeability is one of the most important rock properties in deep geological disposal of high-level radioactive waste as crucial data for geological characterization and numerical simulation. Also, detailed geological characterization of the porosity and permeability of potential CO₂ geological storage must be conducted to determine the storage capacity and injectivity. The directional permeability of rocks for deep geological disposal determines its disposed high-level radioactive waste migration behavior and leakage risk. Direct and indirect methods of permeability measurement have been examined with various experiments. Direct measurement uses a fluid flowing through a core sample under steady-state and unsteady-state conditions. The disadvantage is that direct measurement requires a very long time to measure the accurate permeability of tight formation rocks. To overcome these disadvantages, the pressure pulse-decay method has been developed for measuring permeability called ‘ultra-low permeability’, using unsteady-state flow through core samples. Indirect measurement is a method to predict permeability using granular porosity, grain size, pore-throat, and pore-size distribution through mercury injection, effective electrical conductivity, etc. However, these indirect methods are less accurate than direct methods because these methods are based on many empirical correlations (Kozeny, 1927; Carman, 1937; Swanson, 1981; Katz and Thompson, 1986).

In addition, it is important to understand permeability enhancement in tight formations with natural or artificial fractures. In this study, the permeability of tight core samples was measured before and after fracture stimulation. The observed evolution of permeability through hydraulic fracturing was examined by considering the mechanical and chemical behavior of formation and fluids due to the effects of secondary dissolution and precipitation of minerals on the fracture surface.

The direct and indirect methods of permeability measurement were described, and the permeability was measured before and after hydraulic fracturing. As a result, the stress-enhanced dissolution of minerals increases the density and reprecipitation of mineral matter further occludes pores together these mechanisms additively reduce porosity and permeability.

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Keywords: Permeability, Directional permeability, Hydraulic conductivity, Fracture conductivity, Deep geological disposal

Impact of Fracture Intersection Characteristics on Solute Transport in Fractured Rock Networks

Dahye Kim

Korea Atomic Energy Research Institute (KAERI)

dahye@kaeri.re.kr

In the geological disposal of high-level radioactive waste, natural barriers must play a role in delaying the migration of radioactive nuclides for long periods. In Korea, low-permeability crystalline rocks are being considered as host rocks, and nuclides migration mainly occurs through relatively permeable rock fractures. When these fractures intersect, fluid and solute redistribution occurs at the fracture intersections, and the geometrical and hydraulic properties of the intersections can determine the flow distribution across the entire fracture network. However, simplified models that disregard these detailed characteristics of the fracture intersections are frequently used for computational efficiency in predicting solute transport in large-scale fracture networks.

This study aims to explore whether the effects of geometric and hydraulic properties at fracture intersections on solute behavior observed in small-scale models hold true in large-scale fractured rock networks. Initial numerical simulations were conducted on small-scale fracture intersections to identify key factors influencing fluid and solute behavior, such as aperture distribution and intersection angle. These factors were then used to validate solute behavior in extensive fracture network models. The findings from this study will enhance the understanding of solute transport in complex subsurface environments, offering insights for developing more accurate theoretical models and practical applications in the context of radioactive waste disposal.

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Keywords: Fractured rock networks, Fracture intersection, Solute transport, Solute redistribution, Numerical simulations

Comprehensive Verification and Validation of APro Near-Field Module for Simulating Radionuclide Transport Considering Coupled Thermal-Hydraulic-Mechanical Processes

Jaewon Lee* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*jwl@kaeri.re.kr

The APro (Adaptive Process-based total system performance assessment framework for a geological disposal system), developed by the Korea Atomic Energy Research Institute (KAERI), is an advanced tool designed to assess the long-term safety of geological radioactive waste disposal systems. One of critical components of this modeling framework is the near-field module, which models the complex interactions of thermal, hydraulic, and mechanical (THM) processes that govern radionuclide transport in geological disposal system.

This study outlines a rigorous verification and validation process for the APro near-field module. The first step involved comprehensive testing against analytical solutions and code comparisons, focusing on various combinations of THM-coupled processes, excluding radionuclide transport. This step is crucial to ensure the module's precision in simulating the intricate THM interactions, such as heat convection, thermal expansion, effective stress change, and evaporation and condensation.

In the next step, the module was further validated to incorporate radionuclide transport under coupled THM conditions. This critical validation step confirms the module's capability to accurately simulate multi-physical interactions including the migration of radionuclides influenced by THM processes in the near-field.

The successful completion of these validation and verification steps has significantly increased the reliability of the APro modeling framework. The validated APro near-field module will be able to contribute to the total system performance assessment of geological disposal system in the future.

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Keywords: High-level radioactive waste repository, Process-based total system performance assessment, APro, APro near-field module, Verification and validation

Long-Term Geochemical Reaction Modeling in the Gas-Groundwater-Bentonite System, Simulating the Buffer Barrier of the HLW Geological Repository Site

Daehyun Shin*, Soyoung Jeon, Soyoung Park, Seon-ok Kim, Wonbeom Heo, and Minhee Lee

Pukyong National University

*d3380@pukyong.ac.kr

The DGR (Deep Geological Repository) is regarded as a viable solution for the permanent disposal of the HLW (High-Level Waste). Ensuring the safe and permanent HLW disposal requires the comprehensive understanding of the hydrogeological evolution processes during the HLW disposal and of the leaked radionuclide reaction behavior in the DGR. The DGR is composed of various barrier systems and the compacted bentonite buffer is considered to be used as a crucial engineering barrier. Characterized by high swelling capacity and low permeability, the bentonite buffer plays an important role in preventing groundwater infiltration and in spreading of radionuclides in the DGR. Throughout the hydrogeological evolution of the DGR, various gases such as H_2 , H_2S and CO_2 can be generated, and these gases can change the properties of the bentonite buffer due to various geochemical reactions.

In this study, we conducted geochemical reaction modeling for generation and dissolution of the active representative gases (CO_2 and H_2S) in the KURT groundwater-gas-bentonite (Bentonil-WRK) system under aerobic (or anaerobic) conditions of the repository. This modeling was conducted by using the PHREEQC version 3.7.3 of a geochemical reaction modeling code, considering the microbial respiration that decomposes organic matter using oxygen and sulfate as electron acceptors as well as the mineral dissolution and precipitation in the bentonite. The modeling was conducted with a time scale that has ranging from 0 to 10,000 years and the long-term stability of the buffer material was assessed by investigating changes in the size of the bentonite buffer and the porosity due to the dissolution of the bentonite and the precipitation of secondary minerals.

The modeling results indicated that after approximately 58 years of the reactions, oxygen was completely depleted, leading to a transition into anaerobic conditions. Subsequently, sulfate was reduced, resulting in the formation of HS^- ions in groundwater. During these processes, it was observed that dolomite and pyrite were precipitated due to the reactions with generated gases. After 10,000 years, an increase in the pore size and a decrease in the size of the bentonite buffer were observed, and this result was attributed to the pH reduction caused by the reactions of generated gases (CO_2 and H_2S). The findings of this study demonstrate the significance of gas-induced geochemical alterations in the evaluation for the long-term safety of the buffer barrier in the HLW repository site.

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Keywords: Bentonite, Buffer, Engineering barrier, Geological repository, HLW

Predicting Radionuclide Behavior in Deep Geological Repository Using Graph Convolutional Networks and Long Short-Term Memorys

Jinuk Lee¹, Dae Seong Jeong², and Kyung Hwa Cho^{2,*}

¹Ulsan National Institute of Science and Technology (UNIST)

²Korea University

*khcho80@korea.ac.kr

To evaluate the safety of engineering barriers in geological repositories, predicting radionuclide behavior in leakage scenarios is crucial. Deep learning models have emerged as a pioneering approach for predicting radionuclide behavior. In this study, radionuclide concentrations in the near field of a geological repository were generated using a process-based reactive transport simulation model. A hybrid deep learning model, which integrates graph convolutional networks and long short-term memory (GCLSTM), was employed to replicate the process-based model results, assessing the applicability of GCLSTM for predicting radionuclide behavior. The GCLSTM model demonstrated a good performance over long prediction horizons. In predicting iodine behavior, the GCLSTM model achieved an average NSE of 0.979 with a prediction length of 60 (approximately 30 months). A change in the radionuclide leakage point led to a performance deterioration, with an NSE decrease of 0.48%. However, the accuracy remained acceptable for practical applications. The quantification of predictive uncertainty further ensured the reliability of the predictions, aiding in the assessment of radioactive waste management strategies. Overall, this study successfully demonstrated the applicability of the GCLSTM model for predicting radionuclide behavior.

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Keywords: GCLSTM, PFLOTRAN, Radionuclide behavior, Prediction, Deep geological repository

A Preliminary Evaluation Study of Radionuclide Release From Reference Spent Nuclear Fuels Using PFLOTRAN Code

Shin Sung Oh¹, Hyung Ju Yun², and Ser Gi Hong^{1,*}

¹Hanyang University

²Korea Radioactive Waste Agency (KORAD)

*hongsergi@hanyang.ac.kr

By 2030, the spent fuel storages in our country are expected to be exceeded their capacities from HANBIT Nuclear. South Korea does not have any interim storage facilities and final disposal system for PWR spent fuels. Consequently, long-term national projects have been conducting for developing technologies for deep geological repository in South Korea. In the safety analysis of the deep geological repository, the accident considering groundwater flows into disposal canister and contacting with spent nuclear fuels is considered as a severe scenario leading to the release of some radioactive nuclides to bio-sphere.

This study considered this accident assuming a canister breach time of 1,000 years after disposal and estimated radionuclide release to groundwater using the PFLOTRAN code. In this analysis, the Instant Release Fractions (IRFs) for the design basis reference PWR spent fuels were specially used. The design basis reference spent nuclear fuels were employed which were selected in our previous study: 16×16 PLUS7 and 17×17 ACE7 fuel assemblies. The 16×16 PLUS7 spent fuel assembly is characterized by 45 MWd/kg, 4.5 wt% initial uranium enrichment, and 436 kg initial uranium loading, while the 17×17 ACE7 one by the same discharge burnup and initial uranium enrichment but 468 kg initial uranium loading. The releases were estimated only for 56 safety-relevant radionuclides for geological repository.

PFLOTRAN4.0, an open-source subsurface flow and reactive transport code was used to calculate radionuclide release fractions. This code has capability to conduct FMDM (Fuel Matrix Degradation Model) which calculates UO₂ matrix dissolution considering a variety of the electro-chemical reactions. In particular, we used the FMDM SURROGATE model which is a machine-learning version of FMDM in PFLOTRAN4.0. After canister breaches, both cladding and disposal canister not only lose their containment performance but also groundwater contacts with the fuel surface. Therefore, radionuclides dissolve quickly in water.

U²³⁸ contributed the most to the nuclide release. The release of Cs¹³⁵ and I¹²⁹ were also relatively high. Radioactivity was highest in the order of Am²⁴¹, Pu²⁴⁰, and Pu²³⁹. These results can be utilized as the foundational data for deep geological repository safety analyses. However, this study did not include considerations such as nuclide solubility limitation.

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Keywords: PFLOTRAN, Fuel Matrix Degradation Model (FMDM), Deep geological repository, Instant Release Fraction (IRF), Spent nuclear fuel, Radionuclide release

Sitting Methodology for Proposed Final Disposal Solution of Spent Nuclear Fuel in South Africa

Refilwe Rebecca Molepo* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*refilwe.molepo@gmail.com

South Africa urgently needs a permanent solution for the disposal of High-Level Waste (HLW) and Spent Fuel (SF), particularly as the Koeberg Nuclear Power Station, in operation since 1984, has recently extended its Unit 1 operating license by an additional 20 years. Unit 2 is also undergoing life extension and refurbishment. The extension of Koeberg's operational life beyond the original 40-year margin necessitates the removal of sufficiently cooled spent fuel from the spent fuel pool to an onsite dry storage facility. However, the spent fuel pool and the onsite dry storage facility reached full capacity in 2018, making the establishment of a HLW disposal site increasingly urgent.

This study aims to investigate the feasibility of constructing a HLW disposal facility within a retired mine in South Africa. The research will provide a qualitative assessment of available mines and identify potential suitable sites for the disposal facility, based on international safety standards, best practice norms, geological characteristics, and lessons learned from overseas experiences. By evaluating sites according to HLW repository siting requirements and leveraging international experience, the study aims to propose a candidate site for the facility.

The research will employ a Multi-Criteria Decision Analysis (MCDA) framework to systematically assess potential sites across South Africa's nine provinces. Three MCDA methods will be applied: SMART (Simple Multi-Attribute Ranking Technique), TOPSIS (Technique for Order Preference by Similarity to Ideal Solution), and AHP (Analytic Hierarchy Process). These methods will be used to rank the sites from most favorable to least favorable based on a set of criteria essential for deep geological disposal of spent fuel, as outlined by industry best practices and International Atomic Energy Agency (IAEA) Safety Guidelines. The study will contrast the results of each MCDA method to identify similarities, differences, and account for uncertainties in weighting factors.

To support this analysis, a survey will be conducted using a database of retired mines in South Africa to select a suitable sample of sites for assessment. These sites will be evaluated against criteria such as population density, infrastructure, proximity to Koeberg and other nuclear facilities, geological stability, hydrology, and safety. Additionally, the study will explore international case studies from countries such as Finland, Sweden, France, Canada, Japan, and South Korea, which have successfully implemented deep geological disposal solutions. The findings will provide relevant stakeholders with critical information to support decision-making processes for the development of a HLW disposal facility in South Africa.

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Keywords: High-Level Waste (HLW), Spent fuel, Geological disposal, Site Selection and Multi-Criteria Decision Analysis (MCDA)

A Preliminary Results for Electrochemical Dissolution Rate of Spent Fuel Uranium With Implementation of Mixed Potential Model

Zafar Iqbal Zafar and Ser Gi Hong*

Hanyang University

*hongsergi@hanyang.ac.kr

A mixed potential model-based calculation methodology for assessing electrochemical dissolution rate of Uranium fuel from spent fuel after contacting to underground water resulted from breach of canister is developed. This dissolution rate is proportional to release rate of the radionuclides and gives a glimpse into the composition of the fluids inside a fuel storage canister. The deep geological repository (DGR) conditions assumed firstly no convection in the bulk region and secondly the fluid composition is primarily determined by the alpha radiations which causes hydrolysis of water. A constant temperature of 25C was assumed in this preliminary work. A set of seven electro-chemical reactions at the fuel surface (out of which only two directly dictate the fuel dissolution) were considered with thirteen chemical reactions in the bulk region. A one-dimensional problem width of 6.0 cm was divided into four macro regions of different compositions. This problem is spatially divided into 500 fine meshes and the corrosion layer is not considered, which will be considered in future. A group of eight chemical species namely UO_2^{2+} , $\text{UO}_2(\text{CO}_3)_2^{2-}$, U(IV) , CO_3^{2-} , Fe^{2+} , $\text{O}_2(\text{aq})$, H_2O_2 , and $\text{H}_2(\text{aq})$ are assumed to be important. Only the last three are produced from the hydrolysis processes.

A Dirichlet boundary condition at a distance of 6.0 cm from the fuel surface with a constant supply of the Fe^{2+} ions and the concentrations of all other species were assumed to be zero on the right side boundary of the problem geometry. The continuity of the mass transfer across the fuel surface constitutes the boundary conditions at the other end (i.e., left boundary) of the 1D problem. The four types of the probable processes and reactions are assumed to cause the concentration change of a chemical specie : (1) hydrolysis of water due to alpha radiations, (2) diffusion process, (3) phase change reactions dominated by the electron donor Fe^{2+} , and (4) precipitation due to super saturation of certain species.

The prevalent electrical potential is calculated iteratively using the bisection method because there is no net electrical current from all the anodic and cathodic reactions. The electrical current responsible for the corrosion process termed 'corrosion current' which is directly proportional to the fuel dissolution rate, is calculated from the prevalent potential using the Tafel equation. This model was numerically analyzed using 10^6 uniform time steps over 10,000 years. The computing time was less than 22 minutes on a single 11th Gen i7 processor.

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Keywords: Deep geological repository, Mixed potential model, Electrochemical dissolution, Tafel equation, Dirichlet boundary condition

Systematic Dependency of Water Retention Capacity of Compacted Bentonite on Mineralogical Composition: A Comparative Study

Jinwoo Kim, Min-Seop Kim, Seok Yoon, and Jin-Seop Kim*

Korea Atomic Energy Research Institute (KAERI)

*kjs@kaeri.re.kr

In deep geological disposal for high-level radioactive waste, engineered barrier systems (EBS) based on compacted bentonite experience coupled thermo-hydro-mechanical interactions. One example is the interplay between suction and saturation during wetting and drying. It is described by water retention behavior and is therefore critical for predicting the long-term performance of EBS. However, despite extensive research on various types of commercial bentonites, it still remains unclear what are the key factors that determine the water retention and sealing performance of bentonites as buffer materials. In this study, the systematic dependencies of the water retention capacity of various compacted bentonites on their geotechnical properties are examined. A series of laboratory measurements were carried out to characterize the water retention curves of Bentonil-WRK compacted to dry densities of 1.4-1.8 Mg/m³. The data were then fitted with the van Genuchten model, and the fitted model parameters were compared with those of other bentonites to establish general trends. Observations on these trends suggest that the air entry value (AEV) of compacted bentonites varies systematically not only with the dry density but with the mineralogical composition. This offers valuable insights for optimizing selection of buffer materials as well as compaction strategies. It is anticipated that the outcome of this study will benefit the design of bentonite-based EBS for deep geological disposal systems.

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Keywords: Water retention behavior, Compacted bentonite, Buffer materials, Air entry value, Sealing performance

3분과

고준위폐기물처분 (POSTER)

Geochemical and Geochronological Study Into the Long-Term Evolution of the KURT Site

Nak Kyu Kim*, Soolim Jung, Ji-Min Choi, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*kimnk@kaeri.re.kr

Countries planning or constructing high-level radioactive waste repositories focus on studying the underlying bedrock to ensure the safety and stability of these sites. Understanding how the Earth's crust has evolved over millions to billions of years is crucial for predicting future geological changes that could impact the safety of these repositories. While the lifespan of the high-level radioactive waste repository may be considered up to a million years, the geological formations in which they are built have histories spanning tens to hundreds of millions of years. This long-term perspective provides essential insights into potential future changes, allowing us to anticipate related to tectonic environment, geochemical alterations, and other geological processes. The Korea Atomic Energy Research Institute (KAERI) has designated a specific research area to develop advanced site characterization technologies, undertaking comprehensive geological studies to better understand the deep geological environment. The KAERI Underground Research Tunnel (KURT), completed in 2006, plays a critical role in this research by offering a controlled environment to study the geological characteristics of the site.

This study aimed to evaluate the geochemical properties and geological evolution of the crystalline bedrock at the KURT site. The study primarily focused on the granitic rocks and mafic to intermediate dykes in the area. By analyzing core samples extracted from four boreholes (GE-1, GE-2, GE-3, and GE-4) at approximately 200 meters depth, we conducted comprehensive geochemical analyses including zircon U-Pb and whole rock Ar-Ar age dating to explore petrogenesis of the rocks. Geochronological analysis revealed that the granitic rocks were formed around 184 Ma during the Early Jurassic, while the dykes were formed around 74 Ma during the Late Cretaceous. Furthermore, trace elements suggest that the rock formations at the KURT site are associated with tectonic settings related to an island arc environment. This study provides valuable insights into the geological evolution of the KURT site's bedrock and serves as a fundamental basis for constructing a deep geological model. Additionally, it is expected to contribute to the demonstration of disposal performance and the long-term evolution of natural barriers.

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Keywords: KURT, Geochemistry, Geochronology, Crystalline rock, Long-term geological evolution

Simulation of the South African Radioactive Waste Transportation Risk Assessment Using RADTRAN 6 Program

Nobahle Nomvuyiseko Lulama Sicwebu* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*nobahle@ymail.com

RADTRAN is an internationally accepted program used to assess the risks associated with the transportation of radioactive materials (RAM). Initially developed by Sandia National Laboratories under the auspices of the U.S. Nuclear Regulatory Commission (NRC) and the Department of Energy, RADTRAN was first introduced in 1977 alongside NUREG-0170, the first environmental statement addressing radioactive material transportation. Over the years, the software has evolved significantly. The latest iteration, RADTRAN 6, released in 2009, builds upon the capabilities of RADTRAN 5 by integrating a user-defined dispersion model, a model for incidents involving lead shielding loss, an alternate ingestion dosage model known as INGEST, and a socio-economic model for analyzing social impact and costs associated with possible RAM transportation risks. These enhancements enable more comprehensive risk assessments, considering various factors that could influence the outcomes of RAM transportation.

An example application of RADTRAN is its use in analyzing the transportation of high-level radioactive waste in South Africa. The waste is transported from the Koeberg Nuclear Power Station (Koeberg) to Vaalputs Centralized Interim Storage Facility. Koeberg, the sole generator of high-level radioactive waste in South Africa, serves as the baseline for this simulation. The transportation route primarily utilizes national road N7, spanning 479.1 km. The transport will utilize the HI-STAR dual-use cask with a long-haul truck, which is monitored under strict limitations set by the South African Road Safety Act, including a maximum speed of 60 km/h.

Notably, the truck will halt at Vredendal Petrol Station for 1.5 hours, an area with an estimated population density of 1,300 individuals per km², which is like the one along the route. In assessing the risk of RAM transportation, it is essential to evaluate the Building Dose Factor (BDF), which quantifies the retention of aerosol particles in ventilation systems, thereby influencing inhalation doses to individuals in urban environments.

Among others, the results of this simulation indicate the dilution factors and deposition velocities, conditional accident probability, in-transit population exposure, possible maximum in-transit individual dose, the number of people under plume footprint in case of a single accident, as well as a detailed analysis of an incident-free scenario.

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Keywords: RADTRAN 6, Simulation, Transportation, High-level radioactive waste, Risk assessment

Modelling Earthquake Scenarios Using GoldSim: Managing Ecosystem Leakage of High-Level Radioactive Waste

Hyun Jin Yu, Ki Tae Yang, and Jong Soon Song*

Chosun University

*jssong@chosun.ac.kr

In domestic high-level radioactive waste disposal facilities, environment barriers and engineered barriers play an important role in preventing the release of radioactive materials to the external environment. However, if the barriers are damaged over time due to environment disasters such as earthquakes or ground subsidence, there is a possibility that radioactive waste may leak into the environment. In particular, the deterioration of the barriers due to environment disasters such as earthquakes must be managed through prediction, which requires scenario modelling using simulation programs.

In this study, GoldSim was used to derive specific earthquake scenarios and modelling radioactive waste management measures. The necessary management measures for the deterioration of various barriers, ranging from engineered concrete barriers to environment barriers, were identified, and the effects of earthquakes on the release of radioactive waste into the ecosystem were modelled based on scenarios.

The modelling results show that earthquakes have a significant impact on the scenarios in which radioactive waste leaks into the ecosystem through disposal containers, silos, concrete barriers and environment barriers. This study highlights the impact of earthquakes on the release of radioactive waste into the environment and suggests ways to optimise radioactive waste management.

In conclusion, scenario modelling using GoldSim can be used as an important tool to ensure the long-term safety of radioactive waste disposal facilities, and can provide an important reference for the design and operation of future disposal facilities. This study will contribute to enhancing the safety of radioactive waste management by sophisticated simulation of various mechanisms related to earthquakes.

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Keywords: High-level radioactive waste, Earthquake scenarios, GoldSim, Barrier deterioration, Ecosystem leakage

Synthesis and Sorption Characteristics of U(IV/VI) Colloids Under Neutral and Weakly Alkaline Conditions

Junhyuk Ha¹, Wansik Cha², Seungmo Yeon³, Jong-Il Yun³, and Jun-Yeop Lee^{1,*}

¹Pusan National University (PNU)

²Korea Atomic Energy Research Institute (KAERI)

³Korea Advanced Institute of Science and Technology (KAIST)

*jylee@pusan.ac.kr

The presence of colloids may induce significant uncertainties in predicting the migration and retardation behavior of radionuclides. Therefore, the colloidal radionuclides were considered as a major factor hindering the reliable safety assessment of high-level waste repository. In the present work, the formation behaviors and sorption characteristics of colloidal U(IV/VI) were investigated by using quantitative analysis method.

In the first series of experiments involving the U(VI)-bentonite pseudo-colloid sample, a suspension of Bentonil-WRK bentonite ($< 63 \mu\text{m}$) was prepared by collecting the supernatant after centrifuging the sample at 2,296 rcf for 30 minutes. The suspension of U(VI)-bentonite was then synthesized by adding 10^{-6} M of U(VI) solution to the bentonite suspension. Subsequently, the pH of the colloidal sample was adjusted to 7.0 using 1 mM HEPES, with the ionic strength maintained at 0.01 M using HCl/NaOH and NaCl. The ICP-MS method was employed for analyzing the adsorption characteristics.

In the second series of experiments involving the intrinsic U(IV) colloid sample, a suspension of U(IV)-silicate was prepared by adding a silicate solution to a U(IV)-carbonate solution. The molar ratio of $[\text{Si}]/[\text{U}]$ was 4, and the pH of the suspension was set to 10.1. The U(IV)-silicate solution was then centrifuged at 2,296 rcf for 30 minutes, and the supernatant was used as the stock U(IV)-silicate suspension. The size distribution, oxidation state, and stability of both U(IV/VI) colloids were analyzed with DLS/ELS, TEM-EDS, and XPS methods. Additionally, the adsorption reaction of free Sr(II) ions onto U(IV)-silicate colloids was evaluated at various concentrations of $[\text{Sr(II)}] = 10^{-5}$ M, 10^{-6} M, and 10^{-7} M. All experiments were conducted under an inert atmosphere.

The results obtained in the present work revealed that the pseudo- and intrinsic-colloidal particles were composed of primary particles with average diameters of 41.0 ± 12.6 nm and 7.5 ± 1.4 nm, respectively, as measured by TEM. Based on the zeta potential results, both U(IV/VI) colloids exhibited moderate stability with a negative surface charge. Furthermore, the presences of U(IV/VI) in the colloidal particles were clearly observed with XPS analyses. The results from the sorption experiment of free Sr(II) ions onto U(IV)-silicate colloids indicated a considerable adsorption reaction (ca. 34% with a liquid-to-solid ratio of 76.9 L/g) at $[\text{Sr(II)}] = 10^{-7}$ M. The characteristics and sorption behavior of the pseudo- and intrinsic-colloidal particles are expected to be helpful in understanding the behavior of colloidal uranium under deep geologic repository conditions.

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Keywords: Uranium, Bentonite, Colloid, Sorption

Development of Linking Algorithms for Modelling the Coupled Thermal-Hydraulic-Mechanical-Chemical Processes in Discontinuous Rock

Saeha Kwon*, Changsoo Lee, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*saehakwon@kaeri.re.kr

Discontinuities are physical and structural breaks observed in rock masses, such as faults, joints, and fractures, which accompany the anisotropy and heterogeneity of thermal, hydraulic, and mechanical properties. In particular, the discontinuities can potentially act as flow paths due to their high permeability compared to intact rock. The hydraulic characteristics of rock containing such discontinuities are determined by coupled processes influenced by thermal and mechanical effects applied to the discontinuities. Additionally, in cases where coupled thermo-hydro-mechanical processes occur over a long period, such as in a geological repository for high-level radioactive waste, chemical reactions based on in-situ geological conditions can also influence the coupled behavior of the discontinuities. Therefore, to assess the long-term hydraulic performance of discontinuous rock, it is essential to consider the coupled Thermal-Hydraulic-Mechanical-Chemical (THMC) processes occurring not only in the rocks but also on discontinuities.

This research aimed to develop the algorithms linking TOUGHREACT, the integral finite difference method for the thermo-hydro-chemical simulation, and 3DEC, the block-based discrete element method for the mechanical simulation of discontinuous media. Using the discontinuous model generated by 3DEC, a sequential coupling module was constructed between THC and M analyses by transferring the data stored in each rock and discontinuity element. Pressure solution and precipitation were conserved as chemical reactions influencing the coupled processes on the discontinuities by controlling the chemically-induced aperture changes. The developed algorithms were preliminarily verified through benchmark simulations regarding the dissolution of a novaculite sample containing a single fracture by comparing time-dependent Si concentration changes under different temperature conditions. Still, additional validation through lab-scale and in-situ scale experimental studies involving a wider variety of rock types and chemical reactions is required to apply the developed algorithms for demonstrating the realistic coupled processes and interaction in the multi-barrier system of a geological repository.

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Keywords: Numerical simulation, Coupled THMC processes, Fractured rock, Discontinuum, Geological repository

Corrosion of Copper Species at Different Temperatures

Minsoo Lee*, Junhyuk Jang, Nakkyu Chae, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*minm@kaeri.re.kr

In the KURT (KAERI Underground Research Tunnel), an aerobic corrosion test was conducted to examine the effects of temperature on three types of copper: as-SKB forged copper, hot rolled copper, and cold spray coating copper. KURT's aerobic groundwater was flowed into three environmental chambers, each set to a specific temperature: 30°C, 70°C, and 90°C. Corrosion cells, which contained copper samples surrounded by Ca-type Bentonil-WRK, were placed in these chambers.

The samples from the 30°C chamber were exposed for approximately 1.4 years, while those from the 70°C and 90°C chambers were exposed for about 2 years. The specimens from the 30°C chamber showed a reddish appearance, whereas those from the 70°C and 90°C chambers had surfaces that darkened significantly to a reddish-brown color. X-ray diffraction (XRD) analysis revealed that the primary component of the corrosion films on the specimen surfaces was cuprite (Cu_2O). Further analysis using focused ion beam (FIB) techniques examined the elemental distribution of the films, and an alpha-step profiler measured the surface roughness after film removal. The corrosion films were removed using a 2.5% HCl aqueous solution, and the corrosion depth was estimated using the weight loss method. The corrosion depth was the lowest at 30°C. However, the corrosion depth at 70°C was greater than at 90°C, which is likely due to the formation of a denser corrosion film at 90°C. Meanwhile, no clear trend in corrosion depth was observed among the different types of copper at varying temperatures. These ongoing tests are expected to provide further useful information in the future.

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Keywords: Corrosion, Bentonite, Temperature, Forging copper, Coating copper, Rolled copper

Evaluation of Mechanical Properties for KJ & Bentonil-WRK Bentonites

Seok Yoon*, Gi-Jun Lee, Deuk Hwan Lee, Min-Seop Kim, and Min-Hyeong Lee

Korea Atomic Energy Research Institute (KAERI)

*syoon@kaeri.re.kr

The bentonite buffer material is a crucial component for disposing of high-level radioactive waste (HLW). The strength value can be important to evaluate integrity and safety performance of the buffer material. In KAERI, Ca-type bentonites such as KJ and Bentonil-WRK bentonites have been investigated. In this study, unconfined compression and tensile tests were conducted for the Bentonil-WRK bentonite. The unconfined compression strength (UCS) and tensile strength (TS) values for Bentonil-WRK bentonites were compared with KJ bentonite under the similar dry density and water content condition. The UCS and TS values for the KJ bentonite showed very high correlation with dry density, not with water content. The UCS and TS value between KJ and Bentonil-WRK bentonites showed very similar trend, and there was not a big difference of UCS and TS values between two bentonites.

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Keywords: KJ & Bentonite-WRK bentonite, Unconfined compression strength, Tensile strength

Measurement of Thermal Conductivity and Specific Heat of Rock Specimen by Transient Divided Bar Method

Seungbeom Choi, Saeha Kwon, Taehyum Kim, Seong-Jun Ha, and Jin-Seop Kim*

Korea Atomic Energy Research Institute (KAERI)

*ksj@kaeri.re.kr

In the repository environment of high-level radioactive waste, the decay heat generated from the waste must be effectively dispersed to prevent thermal impact, which can deteriorate the overall safety of the disposal system. The decay heat is first transmitted through the engineered barrier systems and then to the surrounding rock mass. Therefore, it is essential to design heat dispersion properly and to evaluate the thermal properties of the related media accurately. In many rock engineering projects, thermal properties, such as thermal conductivity, specific heat, thermal diffusivity, and thermal expansion coefficient, are usually considered. Various techniques have been proposed to measure the properties, which can be broadly categorized into static and transient methods. The divided bar method is representative of the static category, which is widely applied to measure thermal conductivity. The conventional divided bar method utilizes temperature values at the thermal equilibrium so that it is only capable of measuring conductivity. A modified method, i.e., transient divided bar method (TDBM), enables to measure thermal conductivity and specific heat simultaneously by measuring absorbed heat under varying temperature conditions. TDBM has the advantages of simultaneous measurement using a single specimen as well as broad versatility since it can accommodate commonly used NX-sized rock specimens. Those advantages can draw experimental results with higher representativeness and correlation than other measurements that use much smaller specimens, which makes it challenging to consider the inherent rock mass heterogeneity.

This study adopted the TDBM to measure the thermal properties of rock specimen, mainly focusing on thermal conductivity and specific heat. This method was selected for its ability to provide comprehensive and relevance data from a single specimen. The rock specimens, mostly crystalline granite rocks, were obtained from KURT (KAERI Underground Research Tunnel), specifically from a designated future demonstration test site. The results showed that the specimens had specific heat ranging from 0.73 to 0.93 J/gK and thermal conductivity ranging from 1.80 to 3.77 W/mK. The measurements provided site characterization results that can be used as fundamental data for the future demonstration test.

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Keywords: Transient divide bar method, Thermal conductivity, Specific heat, Single specimen

Simulation of Fluid Flow in a 3D Discrete Fracture Network Using Hydro-Mechanical Coupled Pore Network and Bonded Particle Model and Application to Repository Site Safety Assessment

Haimeng Shen¹, Anne Strader¹, Jeoung Seok Yoon^{1,*}, Mijin Kwon², and Jeong-Hwan Lee²

¹DynaFrax UG LTD

²Korea Radioactive Waste Agency (KORAD)

*js.yoon@dynafrax.com

In safety assessments for radioactive waste repository, understanding fluid flow within rock masses is critical due to its influence on the transport of radionuclides and overall repository integrity. The presence of fractures significantly affects fluid dynamics and the mechanical behavior of the rock, making it essential to model these systems accurately. This study introduces a new modeling technique that utilizes a bonded particle model (BPM) with hydro-mechanical (HM) coupling through a pore network model to simulate fluid flow in rock masses containing a three-dimensional discrete fracture network (3D DFN). The 3D DFN represents a realistic approximation of the fracture systems encountered in geological formations, capturing the complex interactions between fractures and the surrounding rock matrix. By integrating HM coupling, the model accounts for the dynamic feedback between fluid flow and rock deformation, providing a more accurate prediction of fluid migration pathways and stress evolution within the rock masses. This novel approach offers significant improvements over traditional modeling techniques by better representing the anisotropic and heterogeneous nature of fractured rocks, thereby enhancing the reliability of safety assessments for the radioactive waste repository.

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Keywords: Three-dimensional discrete fracture network (3D DFN), Fluid flow, Bonded Particle Model (BPM), Safety assessment

Oxic Corrosion Model for Copper Canister

Nakkyu Chae¹, Seungjin Seo¹, Richard I. Foster², Jinseop Kim¹, and Sungyeol Choi^{2,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Seoul National University (SNU)

*Choisys7@snu.ac.kr

The corrosion resistance performance of copper canisters in deep geological repositories is one of the most critical issues for licensing such facilities. However, assessing the durability of copper canisters over a span of million years is highly challenging topic and impossible to achieve without computational methods. As a result, all national programs for developing deep geological repository system incorporate modeling methods to evaluate the durability of copper canisters within their respective designs. Unfortunately, South Korea previously lacked a model for evaluating the durability of copper canister against corrosion. To address this gap, we developed a model capable of calculating the corrosion of copper canisters under oxic conditions. Our results indicate that the corrosion depth of the KAERI Reference disposal System can reach up to 9.3 μm , which is comparable to values reported by other national programs, such as 6.4 and 10 μm reported by Sweden and Canada, respectively. Additionally, the oxygen depletion time is estimated to be 2.3 years. These findings suggest that the copper canister would only experience corrosion affecting 0.02% of its 50 mm thickness. Despite these findings, the model must be recalculated in the future, as groundwater conditions may vary depending on the specific site selected for deep geological repository. Nevertheless, this model effectively demonstrates the ability to calculate the corrosion depth of copper canisters under oxic conditions and can be a valuable tool for evaluating the durability of copper canisters in deep geological repositories.

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Keywords: Oxic corrosion, Pyrite oxidation, Aerobic microbial activity, Dissolved oxygen, Deep geological repository

Uniaxial Creep Behavior of Dry and Wet Granite

Seong Jun Ha, Taehyun Kim, Seungbeom Choi, Sae-ha Kwon, and Jin-Seop Kim*

Korea Atomic Energy Research Institute (KAERI)

*kjs@kaeri.re.kr

The safety of deep geological repositories for high-level radioactive waste must be guaranteed for hundreds of thousands of years. To achieve this, understanding the long-term behavior of the host rock mass is essential for accurate safety evaluation. In our study, we investigated the creep behavior of granite—one of the most feasible host rock types under the geological conditions in Korea—by performing uniaxial creep tests. The test material was granite obtained from the KAERI underground research tunnel. We prepared two types of specimens: dry and wet granites in cylindrical shape. Dry specimens were stored in an oven at 105 degrees Celsius for a week before the test, while wet specimens were submerged in water before and during the test. Uniaxial compression tests were conducted to determine the crack damage stress (75% of uniaxial compressive strength) before performing uniaxial creep tests. Then, three creep stresses—corresponding to 80%, 90%, and 100% of the crack damage stress—were used in the creep tests. The tests were terminated either when no significant vertical strain change occurred over a week or when creep failure was observed. The results showed no or minimal change in vertical strain at 80% and 90% of the crack damage stress, while at 100%, the primary, secondary, and tertiary creep stages were clearly identifiable.

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Keywords: Uniaxial creep test, KAERI underground research tunnel, Granite, Uniaxial compression test, Crack damage stress

Numerical Simulation of Illitization of Bentonite Depending on Various Thermal and Chemical Conditions

Taehyun Kim, Jeonghwan Hwang, and Jin-Seop Kim*

Korea Atomic Energy Research Institute (KAERI)

*kjs@kaeri.re.kr

The main goal of the high-level radioactive waste repository is to prevent and delay nuclide migration to the human biosphere, and a general concept of the disposal system includes engineered and natural barrier systems. A natural barrier system indicates the host rock mass, and the engineered barrier system consists of canister, buffer, and backfill materials. Bentonite is the primary material of the buffer components, and swelling is one of the crucial characteristics of a buffer material. Swelling occurs due to the smectite minerals in the bentonite, and the swelling capacity may deteriorate because of the transformation of the smectite to non-expandable illite minerals. Therefore, it is essential to simulate the illitization process in the bentonite under various in-situ conditions when designing the disposal system.

This study used an OGS-PHREEQC numerical simulator to simulate the illitization process, combining OpenGeoSys and PHREEQC based on a sequential non-iterative coupling scheme. OpenGeoSys is an open-source finite element code that solves advection-diffusion equations for flow simulation. Meanwhile, PHREEQC solves the equilibrium and kinetic reactions needed to analyze the illitization process. We modeled a one-directional flow test with a Na-bentonite to verify the illitization process. The illitization process was well-captured, and the amount of illite depended on various temperatures and potassium concentrations. We plan to compare the simulation results with the laboratory test results and add the mechanical effect on the flow and chemical process for realistic coupling simulation.

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Keywords: Bentonite, Illitization, OpenGeoSys, PHREEQC, OGS- PHREEQC

A Comparative Review on Hydrogeochemical Characteristics of Deep Groundwater in Countries Preferring Crystalline Rock for Geological Disposal

Eunhye Kwon*, Jang-Soon Kwon, Kyung-Woo Park, and YeoJin Ju

Korea Atomic Energy Research Institute (KAERI)

*ehkwon@kaeri.re.kr

Crystalline rock is preferred as a host rock for long-term stability of geological disposal in Finland, Sweden, Canada, and South Korea. In Finland, Sweden, and Canada, high-saline groundwater is reported in deep crystalline rocks, which are considered as a suitable environment for disposal facilities. However, in South Korea, groundwater with salinity exceeding that of seawater has not been reported. Therefore, this study reviewed previous studies to identify the causes of groundwater salinity differences among these countries and to compare the disposal environments and radionuclide behavior according to salinity.

Groundwater salinity is classified into fresh water, brackish water, saline water, and brine water based on TDS (Total Dissolved Solids). The average salinity of seawater is reported to be approximately 35‰ and is classified as saline water. The deep groundwaters of Finland, Sweden, and Canada show a wide range of salinities, from brackish water to brine water with salinity higher than seawater. The water quality types also vary from Na-Cl to Ca-Cl. In contrast, deep groundwater of crystalline rock in South Korea shows fresh water and Na-HCO₃ types except for groundwater affected by seawater intrusion in coastal areas. Two hypotheses have been proposed to explain the concentrated salinity: the influence of transgression and evaporation during the Paleozoic era, and the impact of seawater freezing during the Cenozoic glaciations. In addition, the restriction of meteoric water by thick permafrost and ice sheets during the glacial periods, along with water-rock interactions, led to increased salinity and changes in water quality types. However, there are no reports indicating that South Korea was affected by transgressions during the Paleozoic era or glaciers during the Cenozoic era. In summary, differences in geographic location and climatic conditions have led to variations in hydrogeochemical evolution and salinity.

In terms of disposal environment safety, a high-salinity groundwater environment offers the advantage of preventing the intrusion of fresh water due to density differences. Conversely, a low-salinity environment has the advantage of potentially minimizing impacts on the performance degradation of buffers, containers, and structural components. The mobility of radionuclides is also reported to vary according to groundwater salinity. Ultimately, this study highlights the need for South Korea to differentiate from leading disposal countries in evaluating long-term hydrogeochemical evolution and disposal safety.

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Keywords: Geological disposal, Crystalline rock, Groundwater salinity, Transgression, Seawater freezing, Water-rock interaction

A Study on the Determinate Key Factors for Monitoring of High-Level Waste Repository at the Site-Scale Level

Woo Beom Ha, Sang Heon Lee, and Jong Soon Song*

Chosun University

*jssong@chosun.ac.kr

With the expected saturation of on-site spent fuel storage repositories in Korea's nuclear power plants, research is underway to secure a high-level waste repository. Worldwide, research to secure a site for a high-level waste repository is steadily progressing. In Korea, research is currently underway at the national scale, and dissolved oxygen in groundwater, total dissolved solids in groundwater, and hydrogen ion concentration have been identified as the main factors for selecting a site suitable for deep repository disposal. As a result of examining domestic and international data, it was determined that when considering the suitability of the final disposal site, there are limitations in securing the results of post-disposal safety evaluation due to the different geological environments of each candidate site. Therefore, this study compared and analyzed the main factors for safety evaluation of disposal sites in each country and redefined the main factors for the site-scale stage applicable to Korea. The criteria for selecting key factors at the national scale and site scale are as follows. For the national scale, the goal is to ensure that the repository is safe without accidents for 100,000 years, and for the site scale, the goal is to study whether the repository is safe without accidents for 1 million years, and the safety evaluation factors for securing the repository should be derived separately. In addition, we will analyze the criteria for preferred and non-preferred conditions presented in overseas cases to check their applicability in Korea. Based on the results of this study, we will simulate long-term geological environment changes by using hydrogeochemical code programs to evaluate 100,000 years or 1 million years in the long term, and consider radionuclide leakage as a priority among the factors suitable for domestic site selection, and use it as a basis for developing geochemical factor values.

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Keywords: Spent fuel, Geochemical factors, Site-scale level, Repository

Suggestion of Mechanical Excavation Methods for Disposal Tunnel Construction and Deposition Hole Drill

Jung-Woo Cho*, Hong-Seok Lee, and Joo-Young Oh

Korea Institute of Industrial Technology (KITECH)

*chojw1665@kitech.re.kr

Many rock excavation techniques have been proposed for construction of disposal tunnel and deposition hole. The advantages for the methods were not precisely analyzed because the in-situ testing in the disposal have been rarely reported. This paper investigated the mechanical methods for hard rock excavation. The methods' objectivity was categorized in two groups: excavation of disposal tunnel construction and drilling deposition hole. For disposal tunnel, TBM, Roadheader, Reaming and other methods were reviewed and a new rock cutting method was also introduced. For deposition hole, existing drilling methods were briefly introduced and the merit and disadvantage were analyzed. From the analysis, an alternative drilling method was proposed to minimize the excavation damaged zone.

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Keywords: Mechanical method, Rock drilling, Disposal tunnel, Deposition hole, EDZ

Manufacturability Evaluation of Outer Copper Shell for Disposal Canister of Spent Nuclear Fuel Based on Flow Forming Process

Young-Ho Lee*, Jae-Deuk Kim, Jeong-Mok Oh, Jung-Hyun Woo, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*leeyh@kaeri.re.kr

Disposal canister of spent nuclear fuel for deep geological disposal fundamentally consist of a dual-structure design: an inner cast iron insert that plays a structural role and an outer copper shell tailored for corrosion resistance in deep geological disposal environment. The outer copper shell is typically manufactured using rolling and forging processes; however, due to limitations in domestic manufacturing facilities and experiences, alternative methods such as additive manufacturing and flow forming process are being considered. The flow forming process, which forms thick wall materials under compressive pressure conditions to thin wall by applying plastic deformation with inner and outer rollers, is sufficiently adaptable for manufacturing copper disposal canister. In this study, a prototype of the copper shell was fabricated using this flow forming process, and its manufacturability was experimentally evaluated. Microstructural analysis showed that even under severe plastic deformation of both inner and outer surfaces, the copper shell maintained applicable microstructures without any significant defects or cracks due to its high ductility, indicating promising applicability and manufacturability of copper shell as an alternative manufacturing method. For a more detailed evaluation of flow formed copper shell, samples were extracted from this prototype copper shell and subjected to corrosion, tensile, and creep tests, with representative results described herein.

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Keywords: Flow forming, Oxygen-free copper, Disposal canister, Copper shell, Deep geological disposal

A Comparison Between a Vertical Disposal and a Horizontal Disposal for High-Level Radioactive Wastes

Jongyoul Lee*, Heuijoo Choi, Changsoo Lee, Sung-Hoon Ji, and Dongkeun Cho

Korea Atomic Energy Research Institute (KAERI)

*njylee@kaeri.re.kr

There are many host rock formations for the deep geological disposal of high-level radioactive wastes including spent nuclear fuels, such as crystalline rock like a granite, sedimentary rock like a clay and salt rock. Leading countries in the field of deep geological disposal for high-level radioactive wastes have decided the host rock and the disposal type, vertical or horizontal, for disposal by their own geological condition in their land.

In the case of granite, the choice between horizontal or vertical types depends on orientation and the distribution of small fractures that are liable to locally affect the very low permeability of the rock. For example, in the French geological context, the analysis of small fractures has shown that they are most mostly almost vertical, and it was decided to develop the vertical borehole design as a reference, which statistically limits the number of small fractures that could occur at disposal cell walls. On the site where the distribution of the fractures would be rather horizontal, however, a horizontal disposal concept could be adopted using the same design principles of a steel overpack and a swelling clay buffer.

In the case of clay rock, the disposal type proposed for the underground installations is horizontal with little vertical extension. In order to maximize the thickness of the argillites located above and below the repository, the structures are placed in the middle of the formation on a single level. So, in order to minimize the space occupied, horizontal disposal is preferred to vertical one.

For the KBS-3 type disposal in granite rock formation developed by Sweden and Finland, the vertical disposal type (KBS-3V) was decided as a reference concept. But, a horizontal disposal type (KBS-3H) has also been studied to date as a variant of the KBS-3 method and an alternative to the KBS-3V design. KBS-3H is based on horizontal emplacement of supercontainers that contain a canister, bentonite buffer around the canister and a perforated outermost metallic shell in a series in long deposition drifts whereas KBS-3V calls for vertical emplacement of the canister in individual depositions holes bored on the floor of a deposal tunnel. Compared to KBS-3V type repository, in KBS-3H, disposal tunnels are not needed because long horizontal holes in KBS-3H displace both disposal holes and disposal tunnels in KBS-3V. But, KBS-3H is more sensitive to vertical fractures, and needs more research and development on grouting technologies and installation machines for supercontainer etc.

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Keywords: Deep geological disposal, High-level radioactive waste, Vertical disposal, Horizontal disposal, Deposition drift, Supercontainer

A Deep Borehole Disposal Concept for Highly Heat-emitting Radioactive Wastes From Pretreatment Process of Spent Nuclear Fuels as an Alternative Final Management Option

Jongyoul Lee*, Heuijoo Choi, Jungho Hur, Hong Jang, and Hunsuk Im

Korea Atomic Energy Research Institute (KAERI)

*njylee@kaeri.re.kr

For the sustainable use of nuclear power, it is essential to manage the radioactive wastes discharged after the generation of electricity from nuclear power plants in the safest manner and consider all the efficient management options for them. In Korea, where the land area is limited and the amount of high-level radioactive waste (HLW) including spent nuclear fuels to be disposed of is becoming relatively large, it is particularly necessary to maximize the disposal efficiency. According to the basic plan for HLW management announced in 2021 by government, it is considered not only to dispose of spent nuclear fuel directly but also to develop the efficient management technologies. Based on this basic plan, Korea Atomic Energy Research Institute (KAERI) is performing the Development of Optimization Technology for Waste Burden Minimization to reduce the disposal wastes and improve the safety.

In this study, for the wastes with the highly heat-generating radionuclides like Cs and Sr generated from one option of several pretreatment processes, a preliminary deep borehole disposal (DBD) concept as a practicable alternative disposal option was developed. To do this, several DBD technologies such as low-temperature very deep disposal (LTVDD), high-temperature very deep disposal (HTVDD) were reviewed. In addition, DBD reference design for disposal of Cs and Sr from the Hanford Site in USA was examined with waste form, disposal package, borehole emplacement and sealing. Based on the results of these review and examination, potential characteristics such as decay heat, radioactivity and size of the solidified waste form of the highly heat-generating wastes containing the high heat dissipation radionuclides (Cs, Sr) were analyzed. And the concepts of the waste storage can, disposal container for those wastes were designed. With this disposal container, emplacement method in the disposal zone which was the depth from 3 km to 5 km depth in a deep bore hole drilled from surface down to 5 km depth underground was established. Also, for the upper part of the disposal zone which was from 3 km depth to the surface, the sealing method with bentonite, crushed rock and Sic was designed.

The results of this study can be used as an input data to establish the policy for final management option of spent nuclear fuel from nuclear power plants. In the future, more detailed performance assessment including thermal analyses and safety evaluation should be performed with real waste form and disposal site data.

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Keywords: Spent nuclear fuel, HLW, Alternative management option, Pretreatment process, Highly heat generating nuclide, Deep borehole disposal

Identification of Recharge Mechanisms Through Statistical Analysis of Hydrologic Time Series Data at the Unsaturated In-Situ Test Facility

Byeong-Hak Park*, Won-Tak Joun, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*bh-park@kaeri.re.kr

The alteration of surface environments has the potential to compromise the long-term safety of a high-level radioactive waste repository. Thus, gaining a comprehensive understanding of the water cycle involving the atmosphere, surface, and subsurface is crucial to ensure the long-term safety of deep geological disposal systems and to secure public acceptance for the repository. Among the various hydrologic components that form the water cycle, recharge plays a significant role in predicting groundwater flow and transport pathways. This study aims to identify recharge mechanisms by conducting statistical analyses on hydrologic time series data observed at the Unsaturated In-situ Test Facility (UNIT) within the Korea Atomic Energy Research Institute (KAERI). The data includes measurements of rainfall, atmospheric pressure, air temperature, groundwater levels, and groundwater temperatures. Statistical analyses of autocorrelation, cross-correlation, and spectral analysis were performed on this dataset. Prior to applying the Water Table Fluctuation (WTF) method for estimating recharge rates using rainfall and groundwater level time series data, we calculated the effective specific yield of the aquifer based on the relationship between rainfall and groundwater levels. These values were then compared and analyzed using various approaches. Additionally, multi-well and single-well pumping tests were conducted in the shallow boreholes within the UNIT to derive specific yields, which were subsequently compared to enhance the reliability of our estimates. Based on these results, we will first calculate the recharge rates within the UNIT, and then plan to extend this analysis to the area surrounding the KAERI Underground Research Tunnel. The findings from these analyses will contribute to a better understanding of recharge processes and the long-term performance of natural barriers in deep geological disposal systems.

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Keywords: Recharge, Natural barrier, Unsaturated zone, Groundwater monitoring, Time-series analysis, Specific yield

Assessment of Uranium Behavior at Natural Analogue Sites in Korea (Boeun and Geumsan) Using Uranium Isotope

Dong-Hee Kim¹, Ye Ji Kim¹, Jaeyeon Kim¹, Sanghoon Lee², and Kang-Kun Lee^{1,*}

¹Seoul National University (SNU)

²Korea Atomic Energy Research Institute (KAERI)

*kklee@snu.ac.kr

To ensure the safe disposal of high-level radioactive waste, it is essential to demonstrate the long-term safety of the deep geological disposal. In this context, natural analogues that enable the characterization of the behavior of radioactive nuclides over geological time scales have gained attention. Since key radioactive nuclides, such as uranium, can exhibit significant variability in mobility and solubility depending on the groundwater environment, it is necessary to establish indicators to assess their behavior. The disequilibrium of uranium isotopes in groundwater is useful for understanding geochemical conditions, redox states, and groundwater flow paths, but remains underutilized. This study aims to compare the behavior of uranium by analyzing the spatiotemporal distribution of uranium isotopes in groundwater samples collected from the natural analogue sites in Boeun and Geumsan. The analysis revealed that the $^{234}\text{U}/^{238}\text{U}$ activity ratios ranged from 1.27 to 3.80 in Boeun and from 1.44 to 2.64 in Geumsan, indicating deviations from the secular equilibrium of 1. This suggests that the groundwater in Boeun and Geumsan may be undergoing complex geochemical processes. The distribution of $^{234}\text{U}/^{238}\text{U}$ activity ratios and uranium concentrations showed that Boeun mainly lies within the ‘normal reduced’ and ‘down-dip from forming’ areas, while Geumsan exhibited a broader distribution pattern. In Boeun, the $^{234}\text{U}/^{238}\text{U}$ activity ratio increased with depth during both dry and wet seasons. This is likely because reducing conditions dominate at greater depths, limiting uranium leaching while encouraging the preferential leaching of ^{234}U due to the alpha recoil process. In contrast, there is no clear correlation between depth and $^{234}\text{U}/^{238}\text{U}$ activity ratio in Geumsan. Thus, this study suggests that uranium isotopes can be utilized to understand the behavior of radioactive nuclides in natural analogue sites and may serve as fundamental data for evaluating the safety of deep geological disposal.

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Keywords: Natural analogue, Uranium behaviors, Uranium isotopes, Groundwater

Development of a Machine Learning Based Surrogate Model to Predict Physical Phenomena Around Disposal Sites

Gil-Eon Jeong^{1,*}, Soobin Kim², Hong Jang², and Jung-Woo Kim²

¹Hanbat National University

²Korea Atomic Energy Research Institute (KAERI)

*gejeong@hanbat.ac.kr

Conducting Total System Performance Assessments (TSPA) for disposal systems, which involve thousands of disposal holes containing spent nuclear fuel and their geological surroundings over spans of more than 100,000 years, is a challenging task due to the extensive computational resources required for numerical analysis. The most computationally intensive aspect of TSPA is the area surrounding the disposal holes, where various physical processes, including thermal, hydraulic, mechanical, and chemical phenomena, have to be considered. In this research, we address this challenge by developing a machine learning-based surrogate model to efficiently reduce computational effort. The core principle of our approach involves removing the meshes inside the disposal holes and using the machine learning based surrogate model to predict and replace the physics not captured by the simplified model. As a result, our study significantly reduces the computational resources required for TSPA of disposal systems while simultaneously preserving accuracy.

Keywords: Disposal system, Surrogate modeling, Machine learning

Some Parametric Sensitivity Studies for Safety of a Double-layered SF Repository

Youn-Myoung Lee*, Changsoo Lee, and Dong-Keun Cho

Korea Atomic Energy Research Institute (KAERI)

*ymlee@kaeri.re.kr

A double-layered system for disposal of spent nuclear fuels (SFs) has been investigating as another alternative option, which is expected to maximize disposal capacity within the minimum footprint disposal area. The disposal concept is quite straightforward as a double layer extension of the same type of single layered KRS concept. Through this study, a conceptual double-layered KBS-3 concepted repository, in both the KRS type layers of which PWR SF canister is disposed of into the disposition holes under the bottom of the tunnel, the surround of which compacted bentonite is buffered, and the remaining part of the tunnel is finally backfilled with a mixture of crushed rock and bentonite, is postulated and then modeled with GoldSim, and preliminarily assessed to investigate its performance as well as some probabilistic sensitivity behavior in view of such principal input parameters as travel length and hydraulic conductivity to the radiological safety.

Among many possible options for consideration of design for high-efficiency repository system for disposal of spent fuels, a double-layered system has also been mainly focused which is expected to maximize disposal capacity with rather already acquired technology. Based on such disposal strategy with a newly designed GoldSim performance assessment methodology, parametric sensitivities of the input might be required to show a reliable long-term safety of the repository.

Use of Analytical Solutions for Evaluation of Multi-Batch Disposal Strategies in Deep Geological Repository

Ji Eon Kim, Samuel Park, Taehoon Park, Dongmin Kim, and Sungyeol Choi*

Seoul National University (SNU)

*chosys7@snu.ac.kr

To complete a sustainable nuclear fuel cycle, high-level radioactive waste including spent nuclear fuels (SNFs) should be isolated from the biosphere for a prolonged period. Deep geological repositories (DGR), which consist of engineered and natural barrier systems, have been considered as an important candidate for the final step of the nuclear fuel cycle. In designing a DGR, the temperature of the buffer material is regulated to prevent degradation in the performance of the engineered barrier system. There are numerous numerical computer codes which perform thermal analyses, often coupled with hydro, mechanical, and/or chemical analyses of DGRs. However, numerical methods have constraints in repetitive or large-scaled calculations owing to the burden of computational costs. This study utilized an analytical method to evaluate the maximum buffer temperature of deep geological repository in large scale. Here, a concept of multi-batch disposal has been proposed, in which multiple types of canisters with different heat generations are emplaced alternatively. Analytical solutions for two- and three-batch disposal were derived from the base solution of single-batch disposal. The solutions were applied to several cases which reflect disposal schedules and the difference in the time of discharge of SNFs from the reactor core. Results show that a two-batch disposal strategy can lower the maximum temperature by 8.2°C when the time of disposal is delayed over 10 years. The maximum temperature is even lower (9.0°C lower than single-batch) when it comes to three-batch disposal. Two- and three-batch disposal strategies can reduce the unit disposal area by 30% and 40% respectively, and thus the overestimation of total disposal area. Whereas a more cautious approach is required when heat generation at the time of disposal is unified (or interval of the time of discharge is same with that of disposal). In addition, statistical uncertainty analysis on the maximum buffer temperature is performed, considering various water contents (1-20wt%) for GTC4 bentonite buffer. Results indicate that widths of 99% confidence interval are below 1°C. This might have been underestimated due to an assumption in analytical solution that heat transfer through buffer material is in equilibrium all the time.

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Keywords: Spent nuclear fuel, Geological disposal, Analytical solution, Thermal analysis, Multi-batch disposal, Uncertainty analysis

Synthesis and Stability of Hydroxylaluminosilicate Colloids Monitored by Multi-Angle Dynamic Light Scattering

Seungwoo Lee¹, Sehwa Bae², Hyeonjin Eun¹, and Jong-Il Yun^{1,*}

¹Korea Advanced Institute of Science and Technology (KAIST)

²Chungnam National University (CNU)

*jiyun@kaist.ac.kr

Hydroxylaluminosilicate (HAS) colloids are kernels of natural aquatic colloids that are ubiquitous in groundwater. Stable colloids can facilitate the mobility of radionuclides through their formation of pseudo-colloids in aquifer systems. Therefore, a mechanistic understanding of the formation and stability of HAS colloids as well as actinide-sorbed HAS colloids would help evaluate the implications of groundwater infiltration in a deep geological repository.

HAS colloids are formed by a dehydration reaction between $\text{Al}(\text{OH})_3$ and mono- $\text{Si}(\text{OH})_4$. There are two types of HAS colloids, HAS_A and HAS_B . In this study, HAS_A and HAS_B colloids were synthesised from mixtures containing $[\text{Si}]:[\text{Al}] = 2:8$ mM and $2:0.5$ mM, respectively. The mixtures were pre-equilibrated at pH 3.0 for a day, and then the formation of HAS colloids was initiated by adjusting the pH to 5–6. The multi-angle dynamic light scattering (MADLS) was used to determine the particle size distribution (PSD) and the total particle number concentration before and during the formation of HAS colloids.

After pre-equilibration at pH 3.0, the initial particle number concentrations were $6 \times 10^7 \text{ mL}^{-1}$ for HAS_A and $4 \times 10^7 \text{ mL}^{-1}$ for HAS_B reaction mixtures. Their PSDs were similar with maximum peaks at around 86 nm and 430 nm, deduced as poly- $\text{Si}(\text{OH})_4$ species. After adjusting the pH to 5–6 and resting for 7 days, the particle number concentrations of both mixtures increased drastically by more than 10^8 times to $2 \times 10^{16} \text{ mL}^{-1}$ and $1 \times 10^{17} \text{ mL}^{-1}$, respectively. Moreover, the PSD peak at 86 nm disappeared and a peak at about 2 nm was newly visible for both mixtures. After a few weeks, the PSD peaks of larger particles were developed while peaks of smaller particles disappeared. In addition, the particle number concentration of the two HAS colloid mixtures started to decrease.

The huge increase in particle number concentration after 7 days can be explained by a burst nucleation of $\text{Al}(\text{OH})_3$, followed by the formation of HAS colloids. The disappearance of the PSD peak at 86 nm is due to the exhaustion of mono- $\text{Si}(\text{OH})_4$, which leads to the dissociation of the poly- $\text{Si}(\text{OH})_4$ species. In addition, the overall decrease in particle number concentration after several weeks indicates that HAS colloids have some tendency to aggregate and the formation rate of new HAS particles decrease overtime. Despite some aggregation, the HAS colloids remained stably dispersed and exhibited sufficient colloidal stability for colloid-mediated migration.

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Keywords: Hydroxylaluminosilicate colloid, Pseudo-colloid, Colloidal stability, MADLS

A Novel Field-Based Experimental Design for *In-Situ* Tracer Tests in KURT

Seonyi Namgung*, Yong-Kwon Koh, Ki-Jong Jang, Ja-Young Goo, Sang-Ho Lee, and Jang-Soon Kwon

Korea Atomic Energy Research Institute (KAERI)

*synamgung@kaeri.re.kr

The migration mechanisms of radioactive nuclides in the natural barriers of deep geological repositories are primarily governed by complex geochemical interactions with minerals in rock fractures and groundwater as the nuclides travel through these fractures, following natural groundwater flow. Most previous field-based studies have largely focused on hydrological flow to predict nuclide transport pathways. However, these experiments often utilized artificially high flow rates, typically ranging from several tens to hundreds of milliliters per minute, limiting their ability to simulate natural conditions in disposal environments. In this study, we have designed an experiment to investigate the natural transport of nuclides along fractures in a granite rock at the KAERI Underground Research Tunnel (KURT). Preliminary site characterization has been conducted using borehole image processing system (BIPS) and borehole televiewer (BHTV) logging to map fracture networks, including their locations and aperture size. Hydraulic tests have also been conducted to evaluate fracture connectivity. In addition, a novel packer system was designed to monitor real-time groundwater quality and nuclide concentrations under natural flow conditions.

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Keywords: Nuclide migration, *In-situ* tracer test, Natural groundwater flow, Fracture networks, Site characterization

Watershed-Scale Radionuclide Transport and Hydrological Simulation Using Hydrogeosphere

Jinmyeong Lee^{1,*}, Minjeong Kim², Heewon Jeong¹, Kyung Hwa Cho¹, and Jung-Woo Kim²

¹Korea University

²Korea Atomic Energy Research Institute (KAERI)

*jmlee@korea.ac.kr

Accurate simulation of groundwater flow is critical for precise modeling of radionuclide transport, as groundwater dynamics directly govern the migration of radionuclides in the subsurface. However, traditional groundwater modeling approaches often rely on simplifying assumptions, such as assuming saturated conditions and unidirectional coupling between surface and subsurface flows. Hydrogeosphere (HGS) stands out as one of the most advanced models capable of simulating unsaturated zones and bidirectional interactions between surface water and groundwater, making it a state-of-the-art tool for such analyses. In this study, we conducted a 60-year simulation of radionuclide transport and hydrological processes within a 4 km by 8 km artificially defined watershed Artificial Disposal System (ADioS), using HGS. The benchmark problem is designed for a total of 60 years, divided into two 30-year intervals. For the first 30-year period, meteorological data from an arbitrarily chosen station, including precipitation and temperature, observed from 1991 to 2020, was used, with sea level fixed at 0 meters. In the second 30-year period, temperature and precipitation were assumed to decrease by 36.40% and 12.01%, respectively, compared to the first period, with a dropped sea level of 35 meters. Hydrological analysis revealed that surface water and groundwater levels were highly sensitive to precipitation variations, with lower levels observed in the second 30-year period compared to the first. Surface water levels were primarily influenced by groundwater discharge, leading to higher radionuclide concentrations in rivers, lake, and the sea. Although the 60-year simulation of the defined benchmark problem was successfully completed, a significant challenge was the lack of real-world observational data to validate the model outputs, due to the artificial nature of the watershed. Consequently, future research will focus on applying this modeling approach to real-world watersheds. Another challenge was the considerable computational time required for these detailed simulations. Despite the relatively modest size of the 4 km by 8 km watershed, the 60-year analysis took approximately 25 days to complete. This substantial time cost poses a significant barrier to long-term modeling efforts, highlighting the need for future research to explore the integration of machine learning techniques to mitigate these computational demands.

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Keywords: Hydrogeosphere, Numerical model, Nuclide transport, Machine learning

Evaluating Groundwater Age and Subsurface Conditions at the KURT Site Using ^3H and ^{14}C Tracers

YeoJin Ju^{1,*}, Kyung-Woo Park¹, Eunhye Kwon¹, Dugin Kaown², Seong Chun Jun³, Jiwon Park³, and
Kang-Kun Lee²

¹Korea Atomic Energy Research Institute (KAERI)

²Seoul National University (SNU)

³Geogreen21 Co., Ltd.

*yju@kaeri.re.kr

Age dating provides insights into how long groundwater has been in a particular location, known as residence time. This information is essential for understanding the natural protection offered by geological barriers over long periods. At the KAERI Underground Research Tunnel (KURT) site in Daejeon, South Korea, isotopic studies were conducted to estimate subsurface hydraulic conditions within the granite formation. Groundwater samples were classified into four groups based on dissolved oxygen (DO) and tritium (^3H) concentrations: (1) modern oxidizing water (DO > 5 mg/L) recharged from the surface, (2) a mixture of modern water and relatively older groundwater with DO between 2–5 mg/L, (3) relatively old groundwater with DO < 2 mg/L, and (4) old groundwater devoid of ^3H . To accurately date the groundwater, ^{14}C dating was performed on samples unaffected by mixing with younger water. The groundwater ages, corrected for the addition of dead carbon, were estimated using dissolved ^{14}C concentrations and ranged up to 15,007 years at the disposal depth. These findings provide valuable insights into the long-term stability and isolation capabilities of the geological barriers at the KURT site, enhancing our understanding of groundwater flow dynamics and contributing to the safety assessment of deep geological repositories.

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Keywords: Groundwater age, Tritium, C-14, Granite, Fracture system, Calcite

Numerical Simulation of Thermo-Hydraulic-Mechanical Behavior in the Horonobe URL Engineered Barrier System Experiment

Minseop Kim¹, Changsoo Lee^{1,*}, Sugita Yutaka², Jin-seop Kim¹, and Min-Kyung Jeon³

¹Korea Atomic Energy Research Institute (KAERI)

²Japan Atomic Energy Agency (JAEA)

³Korea Institute of Geoscience and Mineral Resources (KIGAM)

*leecs@kaeri.re.kr

Assessing the long-term performance of high-level radioactive waste repositories requires coupled relationships in numerical simulations, which must be validated by field-experiment data. The Horonobe engineered barrier system (EBS) experiment, conducted by the Japanese Atomic Energy Agency, serves as a full-scale demonstration of EBS performance. Initiated in January 2015 at a depth of 350 meters within the Horonobe Underground Research Laboratory (URL) in Hokkaido, this experiment contributes to the DECOVALEX-2023 project. In this study, a non-isothermal two-phase flow model was implemented to perform thermo-hydraulic-mechanical (THM) simulations on the heterogeneous porous media of the EBS. The numerical results closely aligning with experimental data. Discrepancies between simulation and experimental outcomes are likely attributed to variations in heater operation and the influence of the Excavation Damaged Zone (EDZ) near the test pit.

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Keywords: Non-isothermal two-phase flow, Primary variable, Horonobe URL, Thermo-hydro-mechanical coupling, Bentonite, Numerical simulation, DECOVALEX-2023

Experimental Study on Nuclide Diffusion in Compacted Bentonite Affected by Reactive Gas

Jin-Seok Kim*, Ji-Soo Kim, Seung Yeop Lee, Jang-Soon Kwon, and Sang-Ho Lee

Korea Atomic Energy Research Institute (KAERI)

*jskim84@kaeri.re.kr

Gas can be generated in a repository for high-level radioactive waste by a variety of mechanisms. First, gas can be generated by corrosion of the metal canister that enclose the spent fuel, and the corrosion of the cast iron insert that contain the nuclear fuel rods. This corrosion is characterized by aerobic corrosion, which consumes oxygen in the presence of oxygen in the initial period after closure of the repository, and then as the repository evolves over time, oxygen is consumed and an anoxic environment is formed. In an anaerobic environment, anaerobic corrosion is mainly caused by sulfur dioxide and sulfide, which are continuously introduced by groundwater or generated/influenced by the activity of sulfate-reducing bacteria (SRB), which can continuously cause corrosion of metal canister. Hydrogen gas can be generated by the corrosion of the metal canister, and the generated hydrogen gas can condense at the canister-buffer interface as the buffer swells and the hydraulic conductivity gradually decreases.

An experimental system was designed to simulate a disposal environment 500 meters underground, and a nuclide diffusion experiment was conducted under conditions simulating the end of the metal canister's life, 1,000 years after the repository's closure. Bentonite compacted to a dry density of 1.61 g/cm³ was used to test the diffusion of nuclides through the buffer material of the engineered barrier. First, the compacted bentonite was saturated with a water pressure of 5 MPa, which is equivalent to the water pressure at 500 meters underground. To evaluate the behavior of nuclides diffusing into the bentonite with dissolved CO₂ gas, the saturated bentonite was injected with gas at a pressure of 5 MPa to balance the water pressure. Cesium iodide was used in the experiment and the diffusion reactions were conducted over 10, 20, 30, and 50 days. Cesium, a mostly cationic nuclide, was adsorbed by the bentonite and diffused at a very slow rate. Iodine, an anionic nuclide, moved out of the bentonite easily.

In scenarios where high-pressure gas is generated at the interface of the canister and buffer, dissolved gases can diffuse into the buffer. This can cause cationic nuclides such as cesium and strontium to attach to and immobilize bentonite particles. Anions readily migrate outside the buffer, so the inhibition by the buffer appears to be minimal. In addition, the reaction of new minerals precipitated by the gas to immobilize nuclides requires separate experiments. Further experiments with hydrogen gas will be conducted in the future to study the diffusion of nuclides by gas.

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Keywords: Nuclide diffusion, Gas generation and transport, Dissolved gas diffusion, Compacted bentonite, Deep geological repository

Thermal Conductivity of Backfill With Degrees of Saturation and Void Ratio Using Silica Sand-mixed Bentonil-WRK

Yohan Cha* and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*Yohan@kaeri.re.kr

Spent fuel is isolated by disposal canisters and buffer, which are placed within deposition holes. Backfill is defined as the materials used to fill deposition tunnels above the deposition holes, serving a crucial role in keeping the buffer's position and limiting advective mass transfer. The performance of backfill is evaluated through its swelling pressure and hydraulic conductivity, while strict standards for thermal properties are less emphasized. However, thermal conditions are a significant variable as they influence the hydraulic and mechanical performance of backfill and are necessary for the overall stability assessment of the repository, including the surrounding host rock. In this study, the thermal conductivity characteristics of silica sand-mixed bentonite blocks were evaluated based on the backfill concept that uses sand or blast-mixed bentonite blocks. The independent variables designed to assess the relationship between saturation, porosity, and thermal conductivity—key input parameters for numerical modeling of thermal properties—were water content (9-22% of bentonite), dry density (1.4-2.0 g/cm³), and mixing ratio (silica mass fractions of 50%, 60%, and 70%). The bentonite used in this study was Bentonil-WRK, owned by the Korea Atomic Energy Research Institute. The experimental results showed that the thermal conductivity ranged from a minimum of 0.5 to a maximum of 2.0 W/m·K, increasing proportionally in order of influence with water contents, dry density, and silica mix ratio. A linear relationship between degree of saturation and void ratio derived from water content and the measured thermal conductivity was established, with a coefficient of determination exceeding 90%. This finding is expected to be used as an input parameter for safety assessments, explaining the thermal conditions of backfill in disposal environments.

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Keywords: Deposition tunnel, Backfill, Bentonil-WRK, Thermal conductivity

Assessment of Ca-bentonite Erosion Using a Modified Two-Region Model

Ji-Won Kim* Minhyeong Lee, Chang-Ho Hong, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*jwk@kaeri.re.kr

Bentonite buffer erosion poses a significant challenge to the long-term stability of the engineered barrier system in high-level radioactive waste disposal. The processes of bentonite saturation, swelling, and gelation during erosion involve complex hydro-mechanical-chemical interactions that require precise numerical modeling to accurately predict their effects on the engineered barrier system. Current modeling efforts, such as the dynamic force balance model, KTH model, and the two-region model developed by SKB, have been widely utilized to simulate the erosion behavior of bentonite. However, these models are predominantly based on the characteristics of Na-bentonite, which exhibits different swelling, gelation, and colloidal properties compared to Ca-bentonite.

In this study, the erosion characteristics of Ca-bentonite was modelled using a modified two-region model implemented via COMSOL Multiphysics. Modifications were made to the diffusivity and viscosity parameters of the original two-region model based on literature and the dynamic force balance modelling results for Ca-bentonite. Notably, Ca-bentonite demonstrated lower diffusivity and viscosity as functions of ionic concentration and bentonite volume fraction compared to Na-bentonite. The revised diffusivity and viscosity parameters were integrated into the two-region model and the erosion characteristics of Na- and Ca-bentonites were evaluated for the Korean reference disposal system and laboratory-scale artificial fracture tests.

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Keywords: Engineered barrier system, Bentonite erosion, Two-region model, Numerical modelling, COMSOL multiphysics

A Comparative Study of APro and HydroGeoSphere for Validating Biosphere Hydrological Simulations

Minjeong Kim^{1,*}, Jinmyeong Lee², Jung-Woo Kim¹, and Kyung Hwa Cho²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea University

*mjkim1@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI) has developed the adaptive process-based total system performance assessment framework (APro) for evaluating the safety of a geological disposal system. APro's biosphere module (hereafter referred to as APro) simulates radionuclide transport in the biosphere accounting for changes in surface environments. It determines the spatial and temporal distribution of radionuclides in aquifers and surface water, using concentrations at the geosphere-biosphere interface (GBI) as the source term.

To simulate radionuclide transport, APro requires prior hydrological simulations. It employs the Soil and Water Assessment Tool (SWAT) to estimate surface water head and groundwater recharge rates, reflecting changes in surface environments. These estimates serve as boundary conditions for groundwater flow modeling, where APro utilizes COMSOL's Darcy's law physics to simulate groundwater flow and discharge. Given the complexity of transient hydrological simulations over long time periods, APro employs a snapshot approach, selecting representative periods where hydrological equilibrium is assumed.

This study validates APro's conceptual model and snapshot-based approach by comparing it with HydroGeoSphere (HGS). HGS is a 3D integrated surface-subsurface hydrological model that uses the diffusive equation for surface flow and Richards' equation for subsurface flow. Its high complexity in the physical processes make it an ideal benchmark.

The benchmark problem includes two short snapshots. While HGS conducted a transient analysis, APro applied its snapshot approach. Although there were minor discrepancies in annual results due to different methodologies, the time-averaged or steady-state analysis results from each snapshot were comparable. Additionally, APro demonstrated significantly greater computational efficiency in terms of time and computing resource usage compared to HGS.

In conclusion, this study shows that APro's hydrological modeling results are comparable to those of HGS while offering superior computational efficiency. This makes it well-suited for long-term safety assessment of a geological disposal system. Future work will focus on validating radionuclide transport simulations.

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Keywords: APro, HydroGeoSphere, Validation, Biosphere, Water balance, Groundwater flow

Practical Analysis of Large-Scale Simulation for Deep Geological Repository by Using APro

Seungjin Seo*, Hong Jang, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*ssj1124@kaeri.re.kr

Along with the trend of nuclear power generation, the necessity of a deep geological repository for the increasing spent nuclear fuels has arisen. To evaluate the safety of the deep geological repository, it is necessary to predict the behavior of radionuclides across the disposal system during the safety assessment period of over hundreds of thousands of years. Because the safety assessment should be based on the simulation of multi-physics such as reactive transport of the radionuclides as well as the groundwater flow and heat transfer in the disposal system, there has been significant requirement for both calculation cost and accuracy in performing large-scale simulations.

APro (Adaptive Process-based total system performance assessment framework for a geological disposal system) has been developed to predict the behavior of radionuclides in a deep geological repository using COMSOL Multi-physics software. The APro can simulate the multi-physics such as reactive transport of radionuclides, groundwater flow, heat transfer, and corrosion of canister in the repository considering several long-term scenarios. In order to simulate such a large-scale geological disposal system, APro needs to improve its computational efficiency while maintaining accuracy.

In this study, a practical study for the APro was conducted to estimate the calculation cost and accuracy in the large-scale disposal system and suggest the optimization direction of mesh selection and solver options in the APro in the massive simulation. By using a cloud cluster computing system and parallel computing module in the COMSOL Multi-physics, the calculation cost, memory requirement, and accuracy were compared and analyzed by varying the number of canisters in the repository. In order to reduce uncertain factors in the analysis, simple process, such as non-reactive solute transport in the near field, was considered in this study. Mesh optimization and application of iterative solver in the COMSOL Multiphysics was conducted to reduce the memory requirement during the parallel calculation.

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Keywords: Deep geological repository, Multiphysics simulation, Parallel computing, Calculation cost, Practical improvement

Consideration of Erosion-Deposition in the Surface Environment on Deep Geological Disposal

Nak-Youl Ko* and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*nyko@kaeri.re.kr

There are various factors that lead to long-term changes in the surface environment, including uplift, subsidence, erosion, and deposition. As these factors can also influence the performance and safety of a deep geological repository for high-level radioactive wastes, they are required to be considered and surveyed when selecting a repository site and conducting long-term assessments. Of these factors, erosion may significantly impact the repository by deteriorating isolation performance because it can decrease the distance from the repository to the surface.

On a land area in the surface environment, riverbed erosion is one of the significant erosive factors in the land area. Complex processes, such as weathering and abrasion, cause riverbed erosion. Therefore, some simplified models, including a steam power model, are broadly used to evaluate erosion rate by flowing water in the river. A geomorphic relief contour map or digital elevation model (DEM) is often utilized to provide necessary parameters for models.

Comparing some riverbed erosion models and analyzing their merits and demerits can help find a suitable model for evaluating the riverbed erosion rates of certain surface water and watershed areas interested in a deep geological disposal project. Also, it will be used to evaluate the long-term stability and determine the proper position of a domestic disposal site for deep geological disposal.

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Keywords: Deep geological disposal, Erosion, Surface environment, Long-term change, Riverbed

Evaluation of Swelling and Hydraulic Properties of Various Types of Bentonite for Buffer Applications

Kiwon Kang, Ugyu Jung, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Bentonite is widely utilized in various fields due to its cation exchange capacity, swelling properties, and retention characteristics which are derived from its primary mineral component, montmorillonite. These properties make bentonite particularly suitable as a buffer or backfill material that can prevent the corrosion of disposal containers by blocking the access of groundwater by its self-sealing property and prevent the migration of radionuclides in the event of leakage from disposal containers through its retention properties

In contrast to Sweden and Finland, South Korea has conducted extensive evaluations on the characteristics of domestic bentonite, specifically Kyungju bentonite (Natural Ca-type), as a candidate material. However, considering the continuously increasing amount of spent nuclear fuel and the potential depletion of domestic bentonite resources, recent discussions have emerged regarding the possibility of importing commercial bentonite from foreign countries to secure a large quantity of high-quality bentonite. It is estimated that approximately 635,329 bundles of spent nuclear fuel will be generated in South Korea. When applying the KBS-3 disposal concept developed in Sweden, it is estimated that over 450,000 tons of bentonite (based on a dry density of 1970 kg/m^3) will be required solely for buffer material. Furthermore, the amount of bentonite required will increase significantly when backfill material is also considered.

Therefore, it is essential to select appropriate commercial bentonite as candidate materials by taking into account factors such as safety-related properties, availability, and economic feasibility. Common types of commercial bentonite can be categorized into natural sodium bentonite, natural calcium bentonite, and sodium-activated calcium bentonite. In this study, we measured the basic swelling and hydraulic properties of commercially available bentonites in South Korea, including BDL (natural sodium bentonite), GCL4 (sodium-activated calcium bentonite), WRK, and GBM (natural calcium bentonite).

For the swelling properties, the free swelling index (ml/2g) was evaluated using powder, and the swelling pressure (MPa) was measured using a cell after manufacturing bentonite blocks. In the case of the free swelling index, the swelling index were analyzed in the order of BDL, GTC4, GBM, and WRK. However, for the compressed blocks, it was confirmed that the type of interlayer cations did not significantly influence the swelling properties. The hydraulic properties were derived through the modified constant head permeability test which commonly used for bentonite blocks, and the hydraulic conductivity of the saturated specimens met the generally used criteria (10^{-12}) for all types of bentonites.

Keywords: Buffer, Bentonite, Swelling properties, Hydraulic conductivity

Comparison Analysis of Corrosion Modeling Between HADES and APro for Deep Geological Repository System

Pilhyeon Ju¹, Heejae Ju^{2,*}, Nakkyu Chae², Samuel Park¹, Seongkoo Hong¹, and Sungyeol Choi^{1,*}

¹Seoul National University (SNU)

²Korea Atomic Energy Research Institute (KAERI)

*heejaeju@kaeri.re.kr, choisys7@snu.ac.kr

For long-term storage of spent nuclear fuel in deep geological repository system, the integrity of copper canisters against possible corrosion and failure is one of the main prerequisites that should be satisfied. Various countries such as Canada and Sweden – considering copper canister in their disposal system – try to quantify and calculate the corrosion of canisters in deep geological environments via numerical modeling. Transport-limited Approach and Mixed Potential Theory are two of the specific ways to describe the corrosion behavior; the former applies the conservative assumption that corrosion happens immediately when sulfide arrives at the canister surface, and the latter quantifies the corrosion by calculating the corrosion potential with consideration of electrochemical reactions. As it is necessary to compare the pros and cons of the two ways for effective description of corrosion behavior, Korea Atomic Energy Research Institute (KAERI) developed the simulation code APro which can apply both of these approaches simultaneously.

This research, following the KAERI's development, aims to verify APro with the previously-developed corrosion simulation code HADES from Seoul National University (SNU), and compare the two approaches with the simulation results. KAERI Reference disposal System (KRS) was described in both simulation codes, with simplified conditions of not considering the effect of temperature and the formation of an oxide film with following prevention, but only considering the redox reactions of sulfide and Cu Sulfidation in copper surface, diffusion and advection through Darcy Flow. The simulation results turned out to be similar in both codes; for the maximum corrosion rate, APro derived about 4.60 nm/yr in Transport-limited Approach while HADES calculated it as about 1.51 nm/yr, both at the top corner of canister. In Mixed Potential Theory, APro showed the result about 1.81 nm/yr while HADES gave the value of about 1.51 nm/yr. Regarding the concentration distribution of sulfide, both codes revealed that at the top corner of the canister the concentration had the highest value, where the corrosion depth was calculated to be a maximum value while the side of the canister had the lowest in Mixed Potential Theory. These results verify that both of the codes well describe the corrosion behaviors in the environment of deep geological repository, but further discussion and mesh study will be needed for better understanding and analysis about the results: difference of modeling dimension of 3D (APro) and 2D (HADES), and singularity issue on calculating corrosion rate of APro in Transport-limited Approach.

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Keywords: Copper corrosion, Canister, Simulation model, Numerical analysis, Deep geological repository

Study on the Durability of Monitoring Sensors in High-Level Radioactive Waste Repositories Under Temperature and Radiation Conditions

Sokpheapika Chea¹, Hyun-Joong Hwang¹, Changhee Park¹, Chang-Ho Hong², and Gye-Chun Cho^{1,*}

¹Korea Advanced Institute of Science and Technology (KAIST)

²Korea Atomic Energy Research Institute (KAERI)

*gyechun@kaist.ac.kr

Spent nuclear fuel, a by-product of nuclear energy, emits significant heat and radiation, posing risks to human safety. Deep geological repositories, located 500 to 1,000 meters underground, are considered the most viable solution for high-level radioactive waste (HLW) disposal. However, the heat and radiation from the waste can compromise the structural integrity of these repositories, necessitating continuous structural health monitoring. Piezoelectric sensors, such as accelerometers, acoustic emission (AE), and ultrasonic sensors, are commonly used to detect issues like strength reduction and cracks. Since these storage structures do not allow for sensor replacement once operational, installing sensors that can endure harsh conditions over extended periods is crucial.

To ensure sensor longevity and suitability, we must evaluate their performance under the environmental conditions present in HLW repositories. Typically, an Accelerated Life Test (ALT) determines the relationship between stress and sensor lifespan. In this study, we conducted preliminary experiments to assess the impact of temperature and radiation on monitoring sensors before proceeding with ALT. Sensors were exposed to 103, 309, 515, 721, and 927 kGy radiation doses and then placed in an oven at 170°C. The results demonstrate that combined exposure to radiation and high temperature significantly reduces sensor lifespan, with higher radiation doses leading to quicker failures. These findings highlight the need for further accelerated life tests focused on radiation stress to enhance sensor durability in HLW environments.

Acknowledgements

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Keywords: Monitoring sensor, Temperature, Radiation exposure, HLW repository, Sensor lifespan, Durability

Sensitivity Analysis of the Thermodynamic Properties of the Engineered Barrier System Through Meta Model Development

Hongjeong Lee, Kiwon Kang, Chohyun Park, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Currently, Korea is considering a disposal system based on Sweden's KBS-3 model to dispose of high-level waste. The disposal system uses a multi-barrier concept to protect high-level waste with canister, buffer, backfill, and natural rock. In high-level waste disposal facilities, spent nuclear fuel is placed in canisters for long-term disposal. During this process, the decay heat from the spent nuclear fuel spreads throughout the facility. This decay heat permeates into the engineered barrier system (EBS), and the increased temperature in the EBS influences various chemical reactions. Efficient management of the decay heat from spent nuclear fuel is crucial, and the most effective approach is to control the thermodynamic properties of the engineered barrier system (EBS), which is in close contact with the spent fuel. It is necessary to identify the key thermodynamic parameters of the EBS that most significantly affect the temperature in the disposal facility and to prioritize research or design based on their importance. In this study, the temperature of the disposal facility was modeled using PFLOTRAN, a numerical simulation tool, and based on the temperature data obtained, a meta-model was developed to perform sensitivity analysis of the thermodynamic parameters of the engineered barrier system.

Keywords: Engineered barrier systems, Temperature, High level radioactive waste, Disposal, Pflotran

Diffusivity of U(VI), Sr(II), and Cs(I) in an Altered Granite From KURT

Hyun-Kyu Lee*, Jae-Kwang Lee, Seonggyu Choi, Jang-Soon Kwon, and Sang-Ho Lee

Korea Atomic Energy Research Institute (KAERI)

*hklee@kaeri.re.kr

The safe disposal of high-level radioactive waste (HLW) is essential for sustainable use of nuclear energy, and for this reason, research on deep geological disposal of HLW has been actively conducted. A deep geological repository (DGR) consisting of canisters, buffers, backfill, and natural rock is considered the most effective way to isolate HLW from biosphere. In many countries, crystalline rocks, such as granite, have been investigated as potential host rock due to its low permeability, high strength, and good integrity. Since the host rock is the final barrier for the isolation of HLW, it is essential to investigate the behavior of radionuclides in the host rock to demonstrate the safety of the DGR. This study aims at investigating the diffusion characteristics of U(VI), Sr(II), and Cs(I) in an altered granite from Korea atomic energy research institute (KAERI) Underground Research Tunnel (KURT). The apparent diffusion coefficients of each nuclide in an altered granite matrix were obtained by conducting through-diffusion experiments under oxidation conditions, and the sorption characteristics of the nuclides were confirmed using field emission electron probe micro analyzer. The results of this study are expected to be utilized for determining the migration and retardation factors of radionuclides in rock matrix for the safety assessment of the DGR system.

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Keywords: Deep geological repository, Granite, Radionuclide, Diffusion, Sorption

Measurement of Diffusion Coefficients of Chloride and Dissolved Oxygen in Compacted Bentonite

Taehyung Park*, Jung-Tae Kim, Minsoo Lee, Chang-Ho Hong, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*tpark@kaeri.re.kr

The diffusion of ions in compacted bentonite plays a crucial role in evaluating the integrity and long-term performance of engineered barrier systems (EBS) for the disposal of high-level nuclear waste. Bentonite, due to its low permeability and high sorption capacity, is considered an ideal buffer material to prevent the migration of radionuclides. This study focuses on measuring the diffusion coefficients of chloride (Cl^-) and dissolved oxygen (O_2), two key ions that significantly impact the corrosion behavior of copper canisters, which are designed to encapsulate nuclear waste for extended periods. Corrosion of copper is a major concern as it affects the canister's long-term stability to isolate the radioactive materials. Using a "through-diffusion" experimental approach, we conducted a series of tests to evaluate the transport properties of these ions in highly compacted Bentonil-WRK having different dry density. The through-diffusion technique allows for precise measurement of ion transport across the bentonite over time, providing a direct assessment of diffusion coefficients. The results of this study reveal key insights into the diffusive behavior of Cl^- and O_2 , both of which are critical in the context of predicting long-term copper corrosion. Our findings indicate that the diffusion rates are influenced by factors such as bentonite density, accessible porosity, and ion interactions within the clay matrix. The diffusion coefficient of chloride was $6.3 \times 10^{-10} \text{ m}^2/\text{s}$ with the Bentonil-WRK having 1.3 g/cm^3 of dry density, where it decreased to $8.2 \times 10^{-11} \text{ m}^2/\text{s}$ for 1.6 g/cm^3 of dry density. These experimentally derived diffusion coefficients are essential for modeling the long-term performance of the EBS, particularly in relation to the potential for corrosion of copper canisters in geological repositories. Ultimately, the data generated from this research contribute to the overall safety assessment and design optimization of nuclear waste disposal systems, ensuring robust isolation of hazardous materials over timescales required for environmental and public safety.

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Keywords: Engineered barrier systems, Copper corrosion, Compacted bentonite, Diffusion coefficient

Analysis of Domestic and Overseas Distribution Coefficients for Radionuclides Sorption on Buffer Material Under Aerobic Condition

Jae-Kwang Lee^{1,*}, Seonggyu Choi¹, Hyun-Kyu Lee¹, Sang-Ho Lee¹, Jang-Soon Kwon¹, Jaeun Kang²,

and Wooyong Um²

¹Korea Atomic Energy Research Institute (KAERI)

²Pohang University of Science and Technology (POSTECH)

*jklee1@kaeri.re.kr

Radionuclides released from deep geological disposal facility can migrate to the surface and eventually affect ecosystem including human beings. Radionuclides released from the disposal canister can first encounter the engineered barriers including buffer and backfill materials. Sorption is one of main retardation mechanisms for the migration of radionuclides in the subsurface environment. We measured and analyzed sorption properties of radionuclides on bentonite as a buffer material under aerobic condition, to incorporate them into the safety assessment for the deep geological disposal of high-level radioactive waste, including spent nuclear fuel. Additionally, we collected and evaluated overseas distribution coefficients for bentonite in aerobic condition. This work included the radionuclides as follows; alkali and alkaline earth metals (Cs, Sr, Ba), lanthanides (Sm), actinides (Ac, Am, Cm, Np, Pa, Pu, Th U), transition elements (Nb, Ni, Pd, Tc, Zr), and others (C, Cl, I, Rn, Se, Sn). For the analysis and evaluation of distribution coefficients, the data from Sweden (SKB), Finland (Posiva), Switzerland (Nagra), and Japan (JAEA) were collected, and we proposed recommended distribution coefficient values by comparing it with domestic measured data.

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Keywords: Bentonite, Distribution coefficient, Deep-geological disposal, Radionuclide, Safety assessment, Sorption

Process Analysis for Generic Safety Case Using K-FEP

Myeongjin Kang, Hyungdae Kim, Gyunyoung Heo, and Kunok Chang*

Kyung Hee University

*kunok.chang@khu.ac.kr

According to the analysis of the Korean Radioactive Waste Society, the saturation of the temporary storage of nuclear power plants is expected sequentially from 2031. Accordingly, the need for high-level radioactive waste disposal facilities has emerged.

To establish a repository for high-level radioactive waste, the performance and safety analysis of the repository must be conducted in compliance with regulatory requirements. Safety analysis requires a collection of arguments and evidence, which the IAEA defines as a ‘Safety case.’ Many countries, such as Finland, Sweden, Japan, the United States, etc., are in the process of licensing disposal facilities using this ‘Safety case’.

In most countries, when constructing a safety case, a report was prepared by analyzing the phenomena occurring within the disposal facility and gathering the supporting evidence, which is referred to as ‘process analysis’ (or ‘phenomenological analysis’).

Process Analysis must specify the conditions under which a phenomenon is analyzed, any uncertainties that the phenomenon may involve, and the data supporting the analysis. In phenomenological analysis, the methods of analysis vary by country as follows:

1. Select a major phenomenon that significantly impacts disposal facilities and analyze their impact.
2. Analyze all possible phenomena in the disposal facilities and their impact.

While constructing the safety case, we developed K-FEP using the NEA IFEP list version 3.0 presented by the NEA. To ensure the comprehensiveness of the safety case, we decided to analyze all possible phenomena and subsequently assess their impact.

Since the disposal site has not yet been confirmed, our Process Analysis focused on developing a generic safety case. The design of the disposal facility was based on the concept of the Posiva KBS-3V system, which is similar to the disposal system currently under consideration in Korea. We conducted a phenomenological analysis to develop a generic safety case based on the KBS-3V concept and previous research on the developed K-FEP and disposal facilities.

Keywords: FEP, Process analysis, State variables, High-level radioactive waste, Disposal, Safety case

Anomaly Detection Method for Accelerometers Using Autoencoder

Se-Oh Kim^{1,*}, Hong-Geun Lee¹, Yeong-Jin Hwang¹, Hong-Jun Kim¹, Jeong-Nam Jeong¹, Seok-Won Lee¹,
Jae-Dam Kim¹, and Chang-Ho Hong²

¹SAE-AN Engineering Corp.

²Korea Atomic Energy Research Institute (KAERI)

*seohkim@sae-an.co.kr

To efficiently detect abnormal conditions in accelerometers, we propose an unsupervised learning method based on an autoencoder. Accelerometers play a crucial role in various industrial and research fields. However, abnormal sensor conditions can lead to degraded system performance and safety issues, making early detection of sensor faults essential. In this study, we present a method that detects abnormal signals by utilizing an autoencoder trained on normal accelerometer signal data. Since the autoencoder trained on normal signals cannot effectively reconstruct abnormal signals, the reconstruction error significantly increases, allowing for the identification of abnormal conditions.

Experimental results confirmed that the autoencoder model could detect abnormal conditions in accelerometers with higher accuracy compared to existing anomaly detection methods. This study is expected to contribute to reducing maintenance costs and improving stability in monitoring systems through early detection of sensor anomalies.

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Keywords: Autoencoder, Accelerometer, Anomaly detection, Signal processing

Validation of the Discrete Fracture-Matrix Model in the APro Far Field Module for Groundwater Flow Simulation

Yong-Min Kim* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*kym@kaeri.re.kr

In the near field of a radioactive waste repository, it is commonly assumed that the properties of engineered barrier systems (EBS) are homogeneous. However, in the far field, the presence of geometrically anisotropic inclusions and highly discontinuous materials, such as large fractures, introduces significant complexity.

Korea Atomic Energy Research Institute (KAERI) has been developing a process-based total system performance assessment model, APro, using COMSOL Multiphysics. APro incorporates the discrete fracture-matrix (DFM) model as one option for simulating groundwater flow in the far field of radioactive waste repositories. This model accounts for groundwater flow in both the discrete fracture network (DFN) and the equivalent continuous porous medium (ECPM). In the DFM model, DFN is represented as lower-dimensional element within the ECPM. For instance, fractures represented as 2D planes are embedded within a 3D porous medium.

Accurately simulating flow and transport requires properly linking dependent variables (e.g., pressure, concentration) between fracture planes and the adjacent porous media. COMSOL's default approach for solving groundwater flow in fractured rock assumes pressure continuity at the interface between fractures and the porous media, meaning that dependent variables are the same across both domains.

In this study, we validated the implementation of the DFM model in APro's far field module by simulating a series of benchmark problems with varying levels of complexity, from simplified geometries with two fractures to cases based on real outcrop data. The numerical results demonstrate that APro's far field module effectively reproduces previously reported solutions, confirming its ability to accurately model the effects of both extensive and embedded fractures within the porous media.

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Keywords: APro, Fracture, Discrete Fracture-Matrix (DFM), Far field, Groundwater flow

Evaluation of pH Alteration Caused by Shotcrete in Engineered Barriers of High-Level Waste Disposal Using PFLOTRAN

Chohyun Park, Hongjeong Lee, Kiwon Kang, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

The Engineered Barrier System (EBS) is essential for ensuring the long-term safety of geological disposal of high-level waste. Shotcrete can be considered as an EBS component contributing to the structural stability of the disposal facility and slowing down groundwater infiltration. However, the alkaline substances leached from the shotcrete can alter the geochemical conditions, potentially inducing physical and chemical changes in the surrounding rock, which may compromise the stability of radioactive waste containment and weaken the integrity of EBS components. Thus, it is essential to quantitatively analyze the alteration of geochemical conditions resulting from shotcrete application to ensure the long-term stability and safety of a high-level waste repository.

In this study, we investigate the impact of alkaline substances from shotcrete on groundwater pH and evaluate their effects on the long-term safety of the disposal system using PFLOTRAN, an open-source, massively parallel subsurface flow and reactive transport code. By simulating fluid flow and geochemical reactions in porous media, we aim to quantify the extent of pH changes and assess their implications for the stability of the engineered barriers.

Keywords: Shotcrete, Engineered barrier system, PFLOTRAN, pH

Numerical Investigation for the Impact of Fractures on Coupled Thermo-Hydro-Mechanical Behaviors of Multi-Barrier Systems in Granite

Kwang-Il Kim*, Changsoo Lee, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*kikim@kaeri.re.kr

Crystalline rocks, such as granite, are considered potential host rocks for deep geological repositories (DGRs) owing to their low permeability, high strength, and stable geological conditions. However, these rocks often contain pre-existing fractures, which serve as predominant pathways for fluid flow and are more susceptible to failure under thermal stress compared to intact rock. This study numerically investigates the impact of fractures on coupled thermo-hydro-mechanical (THM) behaviors of multi-barrier systems. Two numerical models are compared in terms of the temperature, saturation, pressure, and stress changes: one without fractures (Case 1) and one with a pre-existing fracture intersecting two boreholes. In Case 2, groundwater is supplied more rapidly from the far-field rock mass, leading to a faster increase in saturation and fluid pressure in the near-field rock mass and buffer compared to Case 1. This results in accelerated increase in the thermal conductivity of the buffer in Case 2, facilitating faster heat dissipation from the buffer to the surrounding rock mass. Consequently, the heater releases more heat to maintain its pre-determined temperature of 99°C, resulting in a more rapid temperature increase during the initial periods in Case 2 compared to Case 1. In both cases, no fractures are generated in the rock mass. However, plastic shear slip occurs along pre-existing fractures in Case 2 during both excavation and heating stages. During the excavation stage, shear slip is concentrated around the periphery of the two boreholes due to stress relaxation. On the other hand, during the heating stage shear slip is primarily occurs at the midpoint between the two boreholes due to thermal interactions.

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Keywords: Multi-barrier system, Coupled thermo-hydro-mechanical behaviors, Numerical simulation, Fractured rock mass

Immobilization of Iodine Using Phosphate-based Geopolymer Waste Forms

Byoungkwan Kim^{1,*}, Jueun Kim¹, Sultana Afroja², Jae Hohg Kim², Hyojoo Kim¹, Yunjin Choi¹,
and Wooyong Um^{1,*}

¹Pohang University of Science and Technology (POSTECH)

²Korea Advanced Institute of Science and Technology (KAIST)

*kwan928@postech.ac.kr, wooyongum@postech.ac.kr

Phosphate-based geopolymers (P-GP) synthesized using phosphoric acid and amorphous aluminosilicate material is theoretically characterized by having structural positive charges, depending on the reactive P/Al molar ratio. Herein, iodine was immobilized in a P-GP waste form with various reactive P/Al ratios ranging from 1.0 to 1.50, and its physicochemical properties were evaluated. Considering the surface chemistry of P-GP, iodine in the form of iodate was incorporated at 2wt% relative to metakaolin. Overall, the 7-day compressive strength of P-GP waste forms increased proportionally with the P/Al ratios, regardless of the presence of iodine. P-GP waste forms with a P/Al ratio of 1.5 exhibited a compressive strength of 31 MPa in both iodine-containing and iodine-free P-GP waste forms, indicating high enough to meet the waste form acceptance criteria (>3.445 MPa). The XRD results revealed an amorphous hump indicative of the P-GP structure and anatase, as an impurity present in metakaolin, showing no new crystalline phases related to iodine observed. The FT-IR results identified bands associated only with the P-GP structure, consistent with the XRD findings. After a 19-day leaching test, P-GP with P/Al ratios of 1.25 and 1.50 showed iodine release of 4.8% and 2.1%, respectively, indicating over 90% of initial iodine being immobilized efficiently compared to alkali-based geopolymers in which most anions such as iodide and iodate are almost entirely released (>95%). These results suggest that P-GP could be effectively utilized for the immobilization of radioactive oxyanions that are challenging to be stabilized in cement and alkali-activated geopolymer waste forms.

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Keywords: Phosphate-based geopolymer, Waste form, Immobilization, Iodate, Iodine

Review of Hydrothermal Alteration in Bentonite Buffer Under Potential Repository Conditions

Ja-Young Goo^{1,*}, Seonyi Namgung¹, Ki-Jong Jang¹, Yong-Kwon Koh¹, Sang-Ho Lee¹, Jang-Soon Kwon¹,
and Ho Young Jo²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea University

*jygoo@kaeri.re.kr

Bentonite is regarded as a buffer material for the geological disposal of high-level radioactive waste. The long-term behavior of the bentonite buffer can be influenced by the evolution of the repository environment. The heat generated by spent nuclear fuel raises the repository's temperature, potentially impacting the physicochemical and mechanical properties of the bentonite buffer. In this study, the processes and mechanisms of hydrothermal alteration in the bentonite buffer were reviewed. Hydrothermal alteration of bentonite can occur through processes such as illitization, cementation, chloritization, and kaolinitization. These processes are influenced by temperature, fluid chemical composition, the type of bentonite, and the thermal gradient. Changes in swelling and sealing capacities, hydraulic conductivity, and nuclide retention capacity may result from the transformation of montmorillonite in the bentonite buffer. Further comprehensive research on the influence of hydrothermal alteration on the long-term performance of the bentonite buffer, including its nuclide retention and swelling capacities, is needed.

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Keywords: Hydrothermal alteration, Bentonite, Buffer, Performance, Repository

A Proposal to Update the Radioactive Waste Management Indicators Reflecting Recent Development of Major Countries

Kwan Hee Kim¹, Sung Won Jung¹, Dayeong Lim², Gyutaek Kang³, Junho Jeong⁴, Jae Hak Cheong^{1,*},
and Jin Beak Park^{5,*}

¹Kyung Hee University

²Hanyang University

³Sejong University

⁴Gachon University

⁵Technical Research Centre of Finland Ltd.

*jhcheong@khu.ac.kr, jin.park@vtt.fi

To ensure the safe and environmentally sound management of radioactive waste, International Atomic Energy Agency (IAEA) developed the Indicators of Sustainable Development for Radioactive Waste (ISD-RW) according to Agenda 21 of the United Nations Conference on Environment and Development (UNCED) (1992). The initial phase of the indicator is based on the proposed volume of radioactive waste arisings and, hence, reported the limitation to account for both the relative radioactive hazards of different levels and for disadvantages to the countries with more developed nuclear industries. With this regard, in 2001, IAEA proposed a dimensionless ISD-RW that is based on the concept of “sustainability of radioactive waste management”. This concept can address a state of radioactive waste amount that is planned to be disposed of without any increases. In addition to that the amount of waste for final disposal can be accounted for. Preliminary evaluation with the proposed indicator was done (2003) by addressing many nation-wide efforts to enhance the sustainability of radioactive waste management.

This study aims to examine the sustainability of radioactive waste management, evaluating the latest efforts by ISD-RW to uncover new findings and assess the past efforts of significant nations. This research emphasizes identifying the progress that countries have made since 2003 and aims to contribute to the development of future radioactive waste management policies.

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Keywords: Radioactive waste, Indicator, Sustainable development, ISD-RW, Re-evaluation

Total System Performance Assessment of the ADioS Reference Scenario Using APro

Soobin Kim*, Heejae Ju, In-Young Kim, Hong Jang, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*skim1@kaeri.re.kr

Spent nuclear fuel needs to be safely disposed of due to the radiological risks associated with the migration and exposure of radionuclides. Deep geological disposal has been considered as a viable solution for managing spent nuclear fuel. To investigate radionuclide migration in a deep geological disposal system, Korea Atomic Energy Research Institute (KAERI) has developed an Adaptive Process-based total system performance assessment framework (APro) using COMSOL and MATLAB. APro is a finite element method-based model that includes numerical modules for simulating multi-physics phenomena at the disposal system. KAERI applied this framework to a synthetic site called the Artificial Geological Disposal System (ADioS), which encompasses a three-dimensional domain with near-field, far-field, and biosphere components.

In the previous research, KAERI established a comprehensive procedure of Total System Performance Assessment (TSPA) using APro. Long-term evolution scenarios were developed to explore radionuclide migration influenced by the long-term evolution of ADioS. These scenarios, categorized as Reference, Variant, and Disturbance scenarios, qualitatively describe the multi-physics phenomena that impact material properties. Long-term climate change was represented through variations in the surface environment, modeled using six sequential snapshots (SN0 or SNE0, and SN1~5) repeating every 100,000 years after the first glacier cycle.

In this study, we conducted a TSPA for the Reference scenario under the RCP 2.6 climate change scenario over the first glacier cycle (150,000 years). We analyzed radionuclide migration across the near- and far-fields within the type-A of ADioS, using APro integrated model, which combines numerical modules tailored to the Reference scenario. For the near-field simulation, numerical modules such as hydraulic flow, inventory release, solubility, decay, sorption, radionuclide transport, and canister corrosion were employed. For the far-field, hydraulic flow, decay, sorption, and radionuclide transport modules were included. Our results demonstrated satisfactory performance of the TSPA for the Reference scenario using APro. However, numerical errors were detected in the modeling outputs, indicating the need for further study to address these issues and ensure robust scenario analyses.

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Keywords: Geological disposal system, APro, Reference scenario, Climate change, Long-term evolution

A Review on Principal Uncertainty Parameters to be Considered in Probabilistic Safety Assessment for Deep Geological Disposal System in Granite Host Rock

Subin Kim* and Sukhoon Kim

FNC Technology Co., Ltd.

*sbkim@fnctech.com

To resolve domestic issues on spent nuclear fuel, a deep geological repository is under development in accordance with the 2nd basic plan (draft) for high-level radioactive waste management. For the disposal facility, the long-term safety shall be demonstrated through an assessment for compliance with safety objectives (specified as risk and/or exposure dose). In this process, a probabilistic safety assessment must be conducted, incorporating various uncertainty factors inherent in main components of the disposal system, such as natural barrier system (NBS) and engineered barrier system (EBS). Especially, since the host rock, which is the natural barrier of repository, significantly affects release and transport properties of radionuclides, uncertainties related to geological features must be considered as important. Even though the disposal site in Korea has not yet been finalized, granite, which constitutes a substantial portion of the domestic geological formation, is being considered as one of the most probable options for the host rock. Accordingly, in this study, we first reviewed the probabilistic safety assessment performed in the licensing process for deep geological disposal system of spent nuclear fuel in Finland, where the disposal repository will be constructed and operated in the granite. Thereafter, based on review results, we derived principal uncertainty parameters and considerations for the domestic application in the future.

For the operating license application in Finland, POSIVA conducted probabilistic assessments for four scenarios regarding disposal canister failure, each with 1,000 iterations. In these assessments, various epistemic and aleatory uncertainties were considered in relation to granite. The epistemic uncertainty is primarily due to a simplified geosphere model, and the aleatory uncertainty is mainly caused by variability in groundwater flow at the Olkiluoto site. It was observed that uncertainties for the following parameters were typically applied in the Finnish case: (a) Epistemic uncertainty: Effective diffusion coefficient, porosity, Peclet number, and distribution coefficient, (b) Aleatory uncertainty: advective travel time, and geosphere transport resistance.

Similar range and/or distribution of values could be applicable to the domestic facility in the granite host rock. However, the following should be further studied for future application: (a) Identifying the importance of uncertainties, (b) Sampling method of values, and (c) How to scope uncertainty parameters.

It is expected that the results of this study could be directly applied to the probabilistic safety assessment for deciding appropriateness and demonstrating the long-term safety of the specific disposal site in the future.

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Keywords: Probabilistic safety assessment, Granite, Epistemic uncertainty, Aleatory uncertainty, Deep geological repository

Development of Quality Assurance Procedures and Guidelines for the Development, Verification, and Validation of the APro Code

In-Young Kim* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*iykim@kaeri.re.kr

To perform a Total System Performance Assessment (TSPA) that considers the long-term evolution of the Korean Peninsula, it is essential to comprehensively assess the impacts on disposal facility components arising from various phenomena. Existing system-level TSPA models have been simplified by adopting highly conservative assumptions, which limits their ability to fully account for the complex interactions associated with long-term evolutionary phenomena. To overcome these limitations, Korea Atomic Energy Research Institute (KAERI) is developing the Adaptive Process-based Total System Performance Assessment Framework (APro) as a process-level TSPA tool.

To effectively apply the APro code in the design or licensing of actual disposal facilities, it is crucial to ensure the reliability of code development by implementing a rigorous quality assurance process from the initial stages. Accordingly, this study reviewed the regulations related to quality assurance in the domestic nuclear sector and analyzed quality assurance procedures in accordance with KEPIC, the most widely recognized industrial standard. The quality assurance system used in KAERI's research and development projects was also analyzed and summarized, with a focus on key documentation requirements. In anticipation of future international collaborative research involving the APro code, relevant international standards on software quality assurance were also examined.

Based on these findings, KAERI's quality assurance system was adopted as the basis for the APro quality assurance procedure. Subsequently, guidelines for preparing the Software Verification and Validation Report (SVVR) and individual test report guidelines for APro verification and validation test cases were established. Through the procedures developed in this study, quality assurance for APro software development and verification will be conducted, and certification will be issued accordingly.

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Keywords: APro, Quality assurance, Software Verification and Validation Report (SVVR), Literature review

Design of a User Interface for the Adaptive Process-based Total System Performance Assessment Framework (APro)

In-Young Kim* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*iykim@kaeri.re.kr

The Adaptive Process-based Total System Performance Assessment Framework (APro) is currently being developed as a process-level Total System Performance Assessment (TSPA) tool, enabling more detailed evaluations of long-term evolution phenomena in disposal facilities. The APro code categorizes key phenomena into modules, which users can select to simulate either individual performance assessments or TSPA. The APro code has been developed by integrating MATLAB, COMSOL, and PHREEQC. Initially, it utilized Microsoft Excel-based input files; however, to improve computational efficiency and data management, a text-based input deck format has been adopted. While this format offers greater efficiency for developers and experienced users, it presents challenges for novice users. Specifically, users may encounter significant difficulties in generating input files without missing essential data required for each selected module when constructing assessment models.

To mitigate these challenges, a user interface (UI) for the APro code is currently under development. In this study, we conducted a comprehensive analysis of the APro's input files to design an optimal UI data structure. We also identified the essential input data for each module and assessed the interdependencies between modules to establish the relationships between the UI's front-end pages. Our design focused on creating a highly flexible UI that remains functional even if modifications occur in APro's data structure.

Based on the UI design and development strategy established in this study, we plan to complete the development of an initial alpha version of the APro UI for use by developers on individual PCs. In the future, we intend to extend this to a cloud-based APro UI. Additionally, using this UI design as a foundation, we aim to develop linkages with the data management platform PAPIRUS.

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Keywords: APro, User interface, Input data, Data structure

Criticality and Shielding Assessment for Deep Disposal Facilities Based on Horizontal Disposal

Nak Woong Yang, Min Sik Jeon, Ji Hwan Ma, Seok Geun Cho, and Chang-Je Park*

Sejong University

*parkcj@sejong.ac.kr

Due to the high levels of radioactivity and heat generation, spent nuclear fuel requires temporary storage in wet storage pools located within nuclear power plant sites. However, with the saturation of these storage pools anticipated at several sites, including the predicted Kori Unit 1 decommissioned, the establishment of a high-level radioactive waste disposal facility becomes essential. This is expected to be implemented as a deep disposal facility.

The purpose of this study is to design a canister suitable for domestic nuclear fuel types by referencing the STAD canister used in the United States and to evaluate the radiological safety from assessing the criticality and shielding within the disposal facility. The advantage of horizontal disposal is that it eliminates the need for the rotation process required for vertical disposal and minimizing the ground disturbance that would occur with vertical deposition holes. This paper simulates that modeling the canister and disposal facility using the MCNP6 and ORIGEN-S codes.

In the critical evaluation, when the storage canister loaded with new fuel was flooded, the $k_{\text{eff}} = 0.87717$ met the design target even under conservative conditions. The shielding evaluation employed the WH 14×14 type of nuclear fuel, with a cooling period of 10 years. The radiation dose of 0.03174 mSv/h at the top of the cask and 1.00084 mSv/h at the side was determined to be below the threshold of 2 mSv/h on the surface of the spent fuel storage cask. Furthermore, the shielding was assessed at three locations within the tunnel where the four containers were situated. The highest recorded value was 1.6177E-02 mSv/h, which met the design objective of 0.05 mSv/h for the dose rate within the tunnel.

However, it is necessary to evaluate the structural and thermal stability of the disposal canister and optimize the method of disposition canister in the disposal facility because the deformation over time of the disposal canister is not considered. Unlike conventional vertical disposition, this study explains the benefits of horizontal disposal method and presents the possibility of horizontal disposition in future in deep disposal facilities.

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This paper is a result of The Human Resources Development Project for HLW Management hosted by KORAD.

Keywords: Spent nuclear fuel, Deep disposal, Horizontal disposal, Criticality, Shielding

Computational Stability of Module Combinations for Near-Field and Far-Field Domains in APro

Hong Jang*, Yong-Min Kim, Seungjin Seo, Soobin Kim, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*janghong@kaeri.re.kr

APro (Adaptive Process-based total system performance assessment for a geological disposal system) provides a framework for simulating radionuclide transport affected by thermal, hydraulic, mechanical, and geochemical changes in a geological disposal system for spent nuclear fuel. The geological disposal system, which benchmarks the KBS-3V design, is divided into three domains: near-field (canister, borehole, tunnel, buffer, backfill, and surrounding rock), far-field (host rock), and biosphere (soil aquifer and ground surface). APro using COMSOL separates component models for each domain, and integrates them through synchronized boundary conditions. Each component model consists of a combination of ‘modules’, representing key processes to be considered in each component. The near-field domain has 12 modules for radionuclide/chemical transport, groundwater flow, heat transfer, and degradation of safety function. The far-field and biosphere domains have 9 and 5 modules, respectively, for radionuclide transport, groundwater flow, and long-term evolution.

This study conducts a case study to examine the computational stability of the numerical model, focusing on module combinations for the near-field and far-field domains. Test cases with different module combinations are defined to assess the convergence of the computation and the validity of the results. The module combinations are expanded step by step, addressing computational stability issues by adjusting the model, boundary condition, mesh, and solver settings. Specifically, the basic module combination, which includes radionuclide transport, hydraulic, decay, sorption and solubility modules, is examined and applied to the DECOVALEX-2027 Task F’s Reference Case.

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Keywords: APro, Process-based total system performance assessment, Numerical model, Near-field domain, Far-field domain, Radionuclide transport

Parameter Sensitivity Analysis for the Fracture Transport Benchmark Case With Matrix Diffusion in DECOVALEX-2027

Hong Jang*, Yong-Min Kim, Soobin Kim, Seungjin Seo, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*janghong@kaeri.re.kr

APro (Adaptive Process-based total system performance assessment framework for a geological disposal system) incorporates numerical analysis models to simulate multi-physics phenomena in a deep geological disposal system. As the complexity of the multi-physics phenomena increases, a large number of parameters are required, each of which can significantly impact model predictions. However, applying precise parameter values across the entire disposal system is not feasible. Instead, parameter ranges are determined through experimental studies based on some assumptions, and a suitable value is chosen for each model parameter. Consequently, uncertainty in the parameters used in APro's numerical analysis model is unavoidable, making sensitivity analysis essential to quantify the impact of the parameter uncertainty on the evaluation results.

Simulating multi-physics phenomena in a disposal system spanning several kilometers and over 100,000 years is computationally intensive. Conducting sensitivity analysis, which involves multiple iterations of simulations, requires efficient methods. One such approach is the sampling-based global sensitivity analysis (GSA) method. In this study, GSA methods such as variance-based sensitivity analysis (VBSA) and the elementary effect test (EET) are applied to a benchmark case of DECOVALEX-2027 Task F, focusing on fracture transport with matrix diffusion, to assess their performance. Once verified, these methods will be applied to the DECOVALEX-2027 Task F Reference Case, which involves a real-scale disposal system, to evaluate their applicability within the APro integrated system.

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Keywords: APro, Process-based total system performance assessment, Global sensitivity analysis, Fracture transport, Matrix diffusion, DECOVALEX-2027

Conceptual Design of GoldSim-based Safety Assessment Model for Scoping Analysis on the KBS-3H Disposal System of Spent Nuclear Fuel

Donghyun Lee* and Sukhoon Kim

FNC Technology Co., Ltd.

*dhl213@fnctech.com

According to the 2nd basic plan (draft) for high-level radioactive waste management, the permanent disposal facility is under development, aiming at commercial operation within about 10 years after completing the on-site demonstration study. As the disposal method, a multiple barrier system (i.e. KBS-3) incorporated in the deep geological repository in Finland is being considered as the most probable option. In the design process of the disposal facility, the safety assessment should be conducted for demonstrating compliance with safety objectives. Since the disposal site is not finalized yet, the scoping analysis should be performed for tentatively determining the principal design parameter. In this paper, we developed conceptual design of the safety assessment model for scoping analysis on the horizontal design alternative KBS-3H using GoldSim.

Radionuclide release and transport from spent nuclear fuel to biosphere was modeled. Considering purpose of the analysis, a single module is modeled by defining each component. The canister designed to accommodate spent nuclear fuel consists of a cast iron insert covered by a copper overpack and is surrounded by a bentonite buffer. This buffer is covered by a perforated shell, referred to as the supercontainer. However, the supercontainer is not included in this analysis since impact on the groundwater flow is not expected to be significant. The excavation damaged zone (EDZ) is defined along interface of buffer with host rock. It is assumed that there is a large fracture beyond the EDZ where the host rock has high hydraulic conductivity. The groundwater is assumed to flow parallel to the canister.

Two scenarios are developed for receptors and the representative person is applied. In the first scenario, the groundwater is fully released into the ocean where receptors engage in activities (e.g. boating, swimming) and consume seafood. In the second scenario, the groundwater is fully released into the well exposing receptors through drinking and irrigation. The biosphere dose conversion factor is separately calculated based on these two scenarios.

Even though the scoping analysis is based on a simple and conservative model, the assessment results shall meet the safety objectives. These analyses could provide valuable insights for identifying risks, issues, and areas that need further research. By incorporating site-specific parameters, accuracy of the assessment model can be significantly improved. This work will guide development of more advanced models, ensuring that the disposal system meet the safety objectives and can safely manage high-level radioactive waste over a long period.

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Keywords: KBS-3H, GoldSim, Scoping analysis, High-level radioactive waste, Disposal facility

Feasibility Test for Radionuclides Sorption by Copper-exchanged Bentonite

Jeong-Yun Jang, Seonggyu Choi, Hyun-Kyu Lee, Jae-Kwang Lee, and Sang-Ho Lee*

Korea Atomic Energy Research Institute (KAERI)

*sangholee@kaeri.re.kr

The sustainable use of nuclear energy necessitates the safe disposal of high-level radioactive waste (HLW). For this reason, extensive research has been actively conducted on deep geological disposal of HLW. Understanding the sorption interaction between bentonite and radionuclides within deep geological disposal environments is crucial for the safety evaluation of disposal systems. As the disposal environment changes over time, copper released from HLW canisters may interact with and replace the ions within the bentonite. This has the potential to impact the sorption capacity of bentonite to retain radionuclides. Therefore, in this study, the ion exchange between copper and interlayer cations in bentonite was confirmed, and the influence of different types of radionuclides on the sorption behavior of Cu-exchanged bentonite was investigated. The findings provide critical foundational data for evaluating the stability of bentonite in deep geological disposal environments. The outcomes derived from this study are anticipated to serve as essential reference material for the design and appraisal of deep geological disposal systems.

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Keywords: Cu-exchanged bentonite, Bentonil-WRK, Radionuclides sorption, Deep geological disposal environment

Development of New Sensor for Measuring Total Pressure in Bentonite Buffer Under HLW Repository Conditions

Chang-Ho Hong¹, Jin-Seop Kim^{1,*}, and Hee-Young You²

¹Korea Atomic Energy Research Institute (KAERI)

²JooShin industrial Co., Ltd.

*kjs@kaeri.re.kr

High-level radioactive waste (HLW) is planned to be secured over 500-1,000 m in depth. The deep geological repository consists of engineered barrier system (EBS) and natural barrier system (NBS). EBS is composed of copper-steel canister, bentonite buffer, bentonite-excavated rock mixed backfill, and low pH concrete plug. The bentonite buffer plays crucial roles in preventing canister from tilting or damaging, releasing decay heat from the spent fuel in canister, and postponing groundwater inflow from the host rock. If groundwater flows rapidly into deposition hole, there might be a piping channel. In contrast, continuous groundwater inflow allows the bentonite to expand and intrude the rock fractures. These two phenomena are referred to as piping and erosion, respectively. Piping and erosion can cause mass transfer in disposal system and it degrades the integrity of the disposal facility. In Korea Atomic Energy Research Institute, bentonite erosion and piping test (BEPT) is planned to characterize the erosion and piping of bentonite in disposal condition. The bentonite size of BEPT is about 200 mm in diameter which is hard to install thermo-hydro-mechanical sensors to evaluate bentonite integrity. In this study, small size total pressure sensor whose diameter is less than 30 mm is newly designed and developed to measure the pressure up to 5 MPa with waterproof. Developed sensors are tested in 7 days for high temperature and high pressure (~5 MPa) and measured pressure from developed sensors coincides with reference pressure sensors. In further study, the pressure sensors will be upgraded to endure higher pressure about 8 MPa which is usual swelling pressure of the bentonite.

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Keywords: Disposal repository, Bentonite, Erosion, Piping, Field test, Pressure sensor

Types of Hydrothermal Alteration Occurred in Jurassic Granitoids Around the KURT Site

Soolim Jung*, Ji-Min Choi, Nak Kyu Kim, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*sljung@kaeri.re.kr

Disposal facilities for high-level radioactive waste should be located to provide isolation from the accessible biosphere for hundreds of thousands to a million years after closure. In the concept of geological disposal, granite is internationally considered a favorable rock-type for maintaining such disposal safety performance. The typical advantage of granite is that it has normally homogeneous and massive body without any deformation, but chemical changes, especially alteration, can occur within the granite. Mineralogical changes of the bedrocks and precipitation of secondary minerals along the fractures due to the alteration can affect the geochemical stability and the behavior of radioactive materials of the disposal site. Thus, understanding the distribution and the mineralogical and geochemical properties of altered structures is necessary for assessing long-term safety of the repository. However, in granite, where stratigraphic information is lacking, it is difficult to detect the preference of structural distribution. Therefore, it is essential to understand the types of alteration which the site experienced first, and then the mode of occurrence and sequential study of each type should be followed as next steps. In this study, we investigated possible alteration structures in Jurassic granitoid of South Korea using the boreholes around KURT area as a test site. Highly altered rocks were distinguished from host granitoids, and their chemical and structural differences were compared. Detailed fracture mineralogical studies performed to identify the type of alteration which the KURT area has been experienced. This is preliminary study for the construction of alteration history and lithological model around KURT area. The results will be applied to describe the low to moderate temperature (below $\sim 400^{\circ}\text{C}$), post-magmatic geological and paleo-hydrogeological evolution around KURT site and thereby contribute to provide input to hydrogeochemical modelling. Finally, the results will be applied to developing a methodology to construct a long-term geological evolution history of the natural barrier.

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Keywords: Geological disposal, Natural barrier system, Long-term evolution, Alteration, Deep crystalline rock

Accelerated Laboratory-Scale Experiment of Illite-Smectite Transformation Using Bentonil-WRK

Mi Rae Kim, Bo Gyeong Park, Youngseuk Keehm, and Hyun Na Kim*

Kongju National University

*hnkim@kongju.ac.kr

Bentonil-WRK is a candidate bentonite for engineered barriers in high-level radioactive waste repositories, primarily composed of smectite clay minerals. Smectite is known to progressively transform into illite at geological temperatures above approximately 70°C to 200°C, forming illite-smectite mixed-layer minerals. During this process, smectite's intrinsic properties, such as high swelling capacity and adsorption, may be altered. Therefore, understanding the illite-smectite transformation mechanisms and developing strategies to mitigate the reaction rate are essential for improving the long-term performance of engineered barriers.

In this study, we conducted accelerated laboratory-scale experiments on the smectite-to-illite transformation using Bentonil-WRK to gain insights into the underlying mechanisms of this reaction. Hydrothermal experiments were performed by varying the reaction temperature (150-330°C) and time (7-42 days), using a 1 M KCl solution to accelerate the reaction. XRD analysis of the powdered samples indicated that smectite gradually transformed into illite-smectite as both the reaction temperature and time increased, with a concurrent decrease in feldspar and silica peak intensities. Notably, at temperatures above 300°C, dissolution of existing minerals occurred alongside the precipitation of orthoclase.

To focus on changes in clay minerals, oriented specimens were prepared and saturated with ethylene glycol for XRD analysis. The results showed that as the reaction progressed, the intensity of the S(001) peak decreased, while the intensities of the I(001)/S(002) and I(003)/S(005) peaks increased. Analysis of illite content using NEWMOD (Moore and Reynolds, 1997) revealed an increase of 30-90% with reaction temperature and 35-84% with reaction time. Compared to previous studies, these results indicate a relatively rapid reaction rate. According to Ferrage et al. (2011), illite content may be overestimated due to non-fixed K ions in the interlayer, necessitating reanalysis through ion exchange. After re-exchange, illite content decreased by 20-30%. This indicates that special attention must be given to avoid overestimation of illite content during re-exchange experiments.

This study successfully synthesized illite-smectite on a laboratory scale within a short time under varying reaction conditions. The results of the re-exchange confirmed that illite content can be overestimated. Furthermore, to better understand the illite-smectite transformation mechanism, future work will involve analyzing chemical compositional changes using EPMA.

Keywords: Bentonil-WRK, Illite-smectite transformation, Bentonite, Engineered barrier systems mixed-layer

Comprehensive Cost Assessment of Spent Nuclear Fuel Disposal

Kwi-lim Lee*, Heui-joo Choi, Jong-youll Lee, Chang-soo Lee, and Dong-keun Cho

Korea Atomic Energy Research Institute (KAERI)

*kllee@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI) is conducting extensive research on the storage and disposal of domestically produced spent nuclear fuel. The Underground Research Tunnel (KURT) in Daejeon, operational since 2007, and the Korea Reference Disposal System (KRS), based on the Swedish KBS-3V concept, are key components of this effort. KAERI is also planning a site-specific underground research laboratory for a deep geological repository.

KAERI is developing various alternative disposal systems to improve the efficiency of spent nuclear fuel storage and disposal. Among these, the multi-layer disposal system has been prioritized for cost evaluation to assess its economic feasibility. This research supports ongoing efforts to manage spent nuclear fuel safely.

The multi-layer disposal concept, selected based on expert evaluations, includes an engineering barrier designed for PWR/CANDU reactors. The disposal container and bentonite buffer block are designed to ensure structural integrity under disposal depth conditions. Disposal costs are categorized into disposal containers, operating and labor costs, buffer materials, backfill materials, excavation, and packaging facilities. Among surface facility costs, the cost of disposal containers is particularly significant.

The CAFE (Cost Assessment of the Final Disposal of Spent Nuclear Fuel in Korea) program, developed using Visual Basic, estimates disposal costs for PWR and CANDU reactors. It offers user-friendly features and diverse transportation scenarios. Using the CAFE program, cost evaluations based on the 9th Basic Plan for Electricity Supply and Demand identified major cost factors as disposal containers, excavation, and labor costs. Future analyses will include the costs of alternative disposal systems such as the multi-layer disposal system.

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Keywords: Cost, Spent nuclear fuel, Disposal system, CAFE

Review of NTI Recommendations for a Framework on Sustainable Radioactive Waste Management and Socially Acceptable Siting of Geological Disposal Facilities

Ji Su Kang and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

According to the International Atomic Energy Agency (IAEA), over 22,000 m³ of high-level waste (HLW) and 460,000 m³ of intermediate-level waste (ILW) had been generated globally as of 2018. Since these wastes remain hazardous for long periods, geological disposal is internationally regarded as the most appropriate solution for their safe, long-term management. However, social and political challenges have delayed the implementation of geological disposal facilities in many countries. This issue not only undermines trust in the nuclear fuel cycle but also poses a threat to international safety and security.

In response, the Nuclear Threat Initiative (NTI) launched the Developing Spent Fuel Strategies project in 2013, creating a network of experts from the Pacific Rim to seek sustainable solutions for radioactive waste management. Building public confidence has emerged as a key challenge, with crucial elements for success including a compelling case for geological disposal, a transparent decision-making process, and a comprehensive legal and regulatory framework. This study presents a common framework for the successful siting and social acceptance of geological disposal facilities, which can be adapted to the specific conditions of each country.

Currently, in Korea, siting is underway for the construction of an underground research laboratory (URL) aimed at developing disposal technologies suitable for the country's geological environment. This involves conducting experiments and studies on the unique characteristics of Korean bedrock and the performance of the Korean disposal system at depths similar to actual disposal levels. Additionally, a special law on HLW management is being drafted, which will require siting once it is enacted. In this process, the recommendations proposed by the NTI can be applied to ensure the safe, secure, and socially acceptable siting of geological disposal facilities for HLW and ILW.

Acknowledgements

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Keywords: Nuclear threat initiative, High-level waste, Long-lived intermediate level waste, Geological disposal facility, Radioactive waste management, Acceptance

Benchmark Verification of the COMSOL-PHREEQC Coupling Scheme in the APro_Reaction Module

Jeonghwan Hwang^{1,*}, Jung-Woo Kim¹, Weon Shik Han², Won Woo Yoon², and Jiyong Lee²

¹Korea Atomic Energy Research Institute (KAERI)

²Yonsei University

*hwangjh@kaeri.re.kr

APro, a process-based total system performance assessment framework, was developed at KAERI to evaluate radionuclide migration, considering coupled THMC (thermal, hydraulic, mechanical, and chemical) processes in deep geological disposal repositories. The APro_Reaction module, one of the computational components of APro, simulates complex geochemical processes, such as the long-term evolution of engineered barrier systems and radionuclide sorption, through the coupling of COMSOL and PHREEQC. The module integrates these two software through an operator splitting scheme, which is further improved by employing an advanced technique that directly invokes PHREEQC within COMSOL. Specifically, the ‘MATLAB function’ feature is used to call PHREEQC, while the ‘Pointwise constraint’ feature in COMSOL is utilized to update the results, enhancing computational efficiency and reducing simulation time.

In this study, several benchmark problems were conducted in a 1D column, based on previous experimental studies, to verify the APro_Reaction module. The first simulation focused on cation exchange of Na, Ca, and Mg at single cation exchange sites within a 0.1 m column, while the second examined Cs cation exchange at multiple cation exchange sites in a 0.15 m column. Additionally, a benchmark simulation for mineral dissolution and precipitation was performed and compared with PFLOTRAN, a code developed by Sandia National Laboratories in the United States.

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Keywords: APro, COMSOL-PHREEQC, Verification, Geochemical reaction, Benchmark problem

Site Characterization Parameters of D-GIVES Using Geophysical Well Logging at Deep Boreholes in Korea

Seong Kon Lee*, Jihun Choi, In Hwa Park, Myung Sun Kim, Kichang Yang, and Dae-Sung Cheon

Korea Institute of Geoscience and Mineral Resources (KIGAM)

*seongkon@kigam.re.kr

It is very critical to set up site assessment factors that fit the geological conditions of each country, and to evaluate suitability of a certain site as a natural barrier for a disposal system based on these assessment factors. Candidate sites should be narrowed down step by step based on these criteria and geoscientific data available. During the period from 2020 to 2024, ten deep boreholes were excavated to a depth of 750 m in order to enhance understanding of the deep geological environment of Rep. Korea. The locations for deep boreholes were selected considering four major bedrock types and tectonic structures. Drilling cores for all boreholes were recovered to analyze geology, rock mechanical properties, and hydrogeological and geochemical investigation such as pumping and injection tests were also carried out to obtain general features of site assessment factors.

Geophysical logging methods are effective tools in that one can obtain properties of deep environment continuously along the borehole and compensate core-based experiments by correlating geophysical well logs with laboratory test results. Geophysical logs generally need recompilation and conversion to acquire suitable site assessment factors, such as dynamic elastic properties, hydraulic parameters of bedrocks. We set up systematic procedures for QA/QC using geophysical logging data to obtain deep geological environment assessment factors that are essential for the site characterization for HLW disposal sites. The geophysical logging data include 3-Arm caliper logging; temperature, EC (electrical conductivity) logging of borehole fluid, and natural gamma logging; gamma-gamma density logging with dual spacings together with borehole diameter logs; normal resistivity logging together with SP (spontaneous potential), single-point resistance, spectral gamma logging (K, U, Th), acoustic televiewer (ATV), and sonic logging with three spacing between a source and 3 receivers. The fracture information is analyzed from amplitude and travel-time of ATV, and P- and S-wave velocities obtained from sonic logging are used to obtain dynamic elastic of rocks.

All primary geophysical data measured by various logging methods as well, as secondary rock mechanical data such as dynamic elastic properties, dip and azimuth of fracture are merged into the database, D-GIVES (Deep-GeoInformation VERification System), for providing deep geo-environment information. We expect D-GIVES will provide fundamental and general site assessment factors of deep geo-environment of Korea, which are necessary for a modeling of long-term safety evaluation of a site.

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Keywords: Geophysical logging, Deep boreholes, Site characterization parameters, D-GIVES

4분과

중·저준위폐기물관리 (ORAL)

Study on Sorption Ability of Glass Fiber Waste in the Domestic Repository and Its Estimated Colloidal Impact

Hyun Woo Song^{1,*}, Moonoh Kim¹, Sang June Park², and Jun-gi Yeom²

¹KONES Corp.

²Korea Hydro & Nuclear Power (KHNP)

*shw@kones21.com

Colloid migration could lead to enhanced radionuclide release from the radwaste repository because radionuclides may sorb onto mobile colloid instead of immobile barriers, thus being drifted along with the colloidal motion. Recently, glass fiber wastes have been found to contain crushed small particles, thus raising the concerns of colloidal impact. This study aims to evaluate disposal safety of glass fiber waste via literature and experimentation.

According to literature, glass fiber dissolution may occur under hyperalkaline environment, like in the cementitious repository. However, under domestic repository condition, where the pH is estimated to achieve about 12.4, glass dissolution may occur quite rapidly. Considering the literature value and experimental result from previous studies, glass fiber waste would completely dissolve after about 30~35 years. Thus, colloidal impact would be unlikely under domestic condition. However, assuming the glass fiber does not completely dissolve under the domestic repository condition, and crushed particles of glass fiber waste floats within the repository, disposal safety was evaluated by evaluating sorption ability of the crushed glass fiber particle.

Under hyperalkaline condition, glass fiber tends to attain negative charges on the surface, attracting the positive ions, thus positive ions may sorb onto glass fiber particles and drift along. However, according to literature, cobalt ions tend to turn into negative ions under hyperalkaline condition, leading to repulsion between the glass fiber particles and cobalt ions. Considering such phenomenon, experiment was conducted to evaluate sorption ability of glass fiber. Experimental solution was prepared to simulate pH and predominant ion concentration of the pore water of the repository. Then, Co, Cs, and Sr was added into the solution to achieve 10 mg/L. Average of 60 mg/L of glass fiber waste was added into 150 mL experimental solution in PTFE containers. Blank samples without glass fiber were also prepared. Then, the samples were separately allocated into heating baths, each set at 95°C and 25°C.

After 1 month, glass fiber was filtered and to ion concentration of the solution was measured. The concentration of solution with glass fiber does not differ meaningfully from the concentration of the blank sample. This means that co, cs, and sr are very unlikely to sorb onto glass fiber under repository condition, thus even when migration of glass fiber particle does occur, colloidal impact would be unlikely. In future, further study is planned to fully understand the detailed colloidal impact of GFWs disposal in the domestic repositories.

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Keywords: Glass fiber, Colloid, Radioactive waste, LILW disposal facility, Safety assessment

Increasing the Concentration Limit for Disposal of Intermediate-Level Radioactive Waste

BeomIn Kim, DongKook Park, Hyosub Kim, and Jaeyong Lee

Korea Radioactive Waste Agency (KORAD)

*bikim@korad.or.kr, dkpark0110@korad.or.kr, hyosubk@korad.or.kr, chungooc@korad.or.kr

The primary goal in radioactive waste management is to limit the impact of radionuclides released from waste on humans and the environment to acceptable levels. This requires selecting appropriate disposal methods based on the physical and chemical characteristics of the waste, and establishing disposal facilities with sufficient depth and robust engineered barriers to safely isolate waste from the biosphere. Additionally, from a regulatory perspective, it is necessary to set and strictly enforce reasonable standards for the radioactivity levels, or disposal concentrations, of waste that can be accepted into disposal facilities.

The classification system for low and intermediate level radioactive waste in South Korea was revised in 2015 to reflect the safety recommendations of the International Atomic Energy Agency (IAEA). The revised system subdivides the previous low and intermediate level waste category into intermediate level waste (ILW), low level waste (LLW), and very low-level waste (VLLW) based on radioactivity levels. It specifies rock cavern disposal for ILW, near surface disposal for LLW, and landfill disposal for VLLW.

For the Gyeongju radioactive waste disposal facility, the 1 phase underground silo facility uses rock cavern disposal, while the Phase 2 facility uses near surface disposal. Consequently, ILW can only be disposed of in the 1 phase underground silo facility. Therefore, it is necessary to derive and regulate appropriate disposal concentration limits for ILW, which requires a comprehensive review from the perspective of disposal safety.

To evaluate and demonstrate the long-term safety of radioactive waste disposal facilities, it is necessary to identify various features, events and processes (FEPs) that could occur in the disposal environment, and based on these, construct and quantitatively analyze scenarios that are likely to affect the performance of the disposal system. Generally, disposal safety assessments are divided into normal scenarios and abnormal scenarios, with the latter encompassing events that, despite their very low probability of occurrence, could cause serious harm to the disposal system if they do occur.

A prime example of such abnormal scenarios is the human intrusion scenario. Human intrusion refers to any human activity that could affect the isolation performance of the disposal facility and cause radiation exposure. Various forms of intrusion activities exist, such as borehole drilling, resource development, and groundwater use, which can cause significant damage to the isolation function of the disposal facility if they occur. However, these human activities involve fundamental uncertainties about the future that are difficult to predict, making their impacts challenging to assess.

Therefore, international recommendations suggest taking a stylized approach based on reasonable assumptions for human intrusion scenarios, rather than strict probabilistic analysis. This involves selecting human activities that are likely to occur and could pose a serious threat to the safety functions of the disposal system, considering the disposal environment and various factors of change. A few stylized scenarios are then constructed and analyzed deterministically using conservative input data and assessment models. The purpose of this assessment is to verify the integrity and safety margin of the disposal facility design by presenting the maximum potential dose constraint due to human intrusion.

Meanwhile, biosphere modeling is important in the disposal safety assessment process. Recent efforts have focused on developing dynamic interaction models that consider biogeochemical cycling to more realistically simulate the dynamic behavior of radionuclides in the environment, overcoming the limitations of traditional equilibrium models.

This study aimed to establish disposal concentration limits for ILW in the 1 phase underground silo facility of the Gyeongju low and intermediate level radioactive waste disposal facility through human intrusion scenario assessment. First, based on an analysis of the characteristics of the post-closure disposal facility and surrounding environment, possible human intrusion scenarios were identified, and representative scenarios were selected according to their likelihood of occurrence and severity of impact. Next, safety assessment methods were applied to the representative human intrusion scenarios to quantify the release of radioactive materials through scenario pathways and the radiation doses to major exposure groups. From this, the exposure dose per unit radioactivity and the upper limit of disposal concentration for each nuclide that satisfies the licensing criteria were calculated. Through this process, disposal concentration limits applicable to the 1 phase underground silo facility were established.

Keywords: Intermediate-Level Waste (ILW), Human intrusion scenarios, Disposal concentration limits

Study for the Possibility of the Penetration Intrusion by Drilling Against 1st Phase Radioactive Waste Disposal Facility

Chan Hong Kim, Hong-Il Kwon, and Beom In Kim

Korea Radioactive Waste Agency (KORAD)

*chahkim@korad.or.kr, hong1k@korad.or.kr, bikim@korad.or.kr

1st phase radioactive waste disposal facility which is running is composed of total 6 silos. Their depths vary between G.L.-127.7 and G.L.-165.0 m. Its life cycle is 300 years and after it'll be shut off.

This study consider human intrusion by drilling after that and the basic assumption is as follows. In the future after many generations, the information including all documents and related persons is disappeared. Furthermore, nobody would remember the existence of silos. At that time, assume the landlord is planning to develop the area on disposal facility. It is obvious to begin with site investigation such as drilling and boring prior to the design and execution of the structure. We ignore the procedure for the approval by local government.

Drilling practice and its final depth depends on the purpose of the site investigation. Therefore, we look up related design code and acts to follow. Otherwise, estimate the possibility of penetration intrusion with valid and acceptable methodology. In this study, we classify the land development into four categories, i.e. housing, infra construction, well development and mining. Final drilling depths are evaluated with the design code directly and the possibility of the intrusion against tunnels is estimated.

Finally, consider the unexpected drilling without any purpose and compare with the details of tunnel structure.

Keywords: Drilling, 1st phase radioactive waste disposal facility, Human intrusion

Advanced Radioisotope Inventory Management for Low- and Intermediate-Level Radwaste in Korean NPPs: Algorithms Improvement and Software Development for Inventory Estimation

Ji Kang Park*, Sung Yeop Jeong, Eui Jin Jeong, and Sang Mo Ham

Godo Tech Co.

*pjk@godo-tech.com

Managing the radioisotope inventory for over 90,000 low- and intermediate-level radwaste drums (based on 200 L) stored at Korean nuclear power plants (NPPs) requires rapid estimation of specific radioactivity by isotope, and the development of software for this purpose. The previous software, developed in 2010, has become outdated, no longer aligning with current disposal conditions. This paper focuses on improving estimation algorithms and developing software with these improvements.

Improving specific radioactivity estimation algorithms by isotope requires different approaches for Easy-to-Measure (ETM) and Hard-to-Measure (DTM) isotopes.

ETM isotopes can be estimated using either the nuclide spectroscopy method or the Dose-to-Curie (DTC) method, but the DTC method is more appropriate for rapid estimation. To improve the DTC method, the ratios of ETM isotopes were calculated using tens of thousands of drum nuclide analysis datasets from Godo Tech, and DTC conversion factors were derived from MCNPX simulations based on the physicochemical data of various drum types and forms.

After estimating ETM isotopes, the DTM isotopes are estimated by applying the scaling factor method to the specific radioactivity values of the ETM isotopes. For radwaste types without developed scaling factors, previous destructive analysis results were used, and values derived similarly to the geometric mean approach of the scaling factor method were applied as scaling factors.

Once the specific radioactivity for all isotopes is estimated, drums are classified according to radioactive level, and disposal eligibility is determined based on criteria.

The software with improved estimation algorithms was developed for user-friendliness and easy upgrades. It minimizes the modification of the drum data format from NPPs and allows estimating multiple drums, with results easily exportable for further processing. Additionally, the software accommodates modifications to fundamental data for specific radioactivity estimation through developer adjustments.

Some data require additional licensed values for disposal, therefore continuous updates to the software and support for these updates are necessary for future radioisotope inventory management.

Keywords: Nuclear power plant, Intermediate & low level radwaste, Radioisotope inventory, Dose to curie, Scaling factor, Radioactivity estimation software

A Study on the Indirect Evaluation Method for Radionuclide Inventory in Past Radioactive Waste

Ji Eun Park*, Wook Sohn, Hang-rae Cho, Seongjin Maeng, and Jungi Yeom

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*Jepark12@khnp.co.kr

For the disposal of low and intermediate-level radioactive waste generated by nuclear power plants, the radionuclide inventory must be evaluated in accordance with the regulations set by the Nuclear Safety and Security Commission. Indirect evaluation methods, such as the scaling factor method, can be employed for this purpose. Scaling factors have been developed for dry active waste (DAW) generated after 2004, and periodic verification is conducted every two years to assess the radionuclide inventory. On the other hand, for waste generated prior to 2004, indirect evaluation methods, such as scaling factors, still need to be developed. However, the direct application of current methods is difficult because clear standards and regulations for packaging were not in place at the time of waste generation.

This study aims to investigate the distribution characteristics of radionuclide concentrations in legacy waste and proposes a new evaluation method that reflects these characteristics. The newly developed indirect evaluation method complies with international standards and incorporates similar cases from overseas. Depending on the correlation coefficients of each radionuclide, the method applies either the scaling factor method or the average activity concentration method. This study provides a detailed explanation of the rationale and statistical techniques used in this approach.

Keywords: Scaling factor, Mean activity concentration, Radionuclide, Inventory

Development of a Scaling Factor Method for PET Cyclotron: Correlation Analysis of Key and DTM Nuclides

Yeijin Bang¹, Byeonghyeon Park², Sangmin Lee², and Hee Seo^{1,*}

¹Jeonbuk National University (JNBU)

²Korea Institute of Nuclear Safety (KINS)

*hseo@jbnu.ac.kr

PET cyclotrons are widely used in production of radiopharmaceuticals, with various nuclear reactions occurring during the operation, causing the activation of the PET cyclotron itself and surrounding materials by protons or scattered particles. Considering the operational lifetime of current domestic PET cyclotrons, the demand for decommissioning is expected to increase gradually, necessitating effective methods for the dismantling and disposal of activated waste. To address this, it is essential to identify correlations between radionuclides in radioactive waste generated by PET cyclotrons and confirm the applicability of the scaling factor method, which is already used in nuclear power plant regulations.

In this study, in order to develop a scaling factor method for PET cyclotron, we evaluated the radioactivity according to the elemental composition and analyzed the correlation between key nuclides (gamma-emitting nuclides) and difficult-to-measure (DTM) nuclides (alpha and beta-emitting nuclides). Based on GE PETTrace 880 model, the evaluation was focused on the havar foil located in the proton beam direction. Also, the beam energy and beam current were set to 16.5 MeV and 40 μ A, respectively, considering H₂¹⁸O liquid target and using the PHITS-DCHAIN simulation code. The composition distribution of the havar foil was determined based on X-ray fluorescence analysis and data from previous studies. After determining the composition distributions, 20 elemental combinations were randomly selected. After a total irradiation time of 125 hours, the nuclide inventory was evaluated following a cooling period of 5 years, taking into account the replacement cycle of the havar foil.

Activation calculations identified a diverse range of radionuclides, including ³H, ¹⁴C, ⁵⁴Mn, ⁵⁵Fe, ⁵⁷Co, ⁶⁰Co, and ⁵⁹Ni. Among these, ⁵⁶Co, ⁵⁸Co, ⁶⁰Co, and ⁵⁴Mn were selected as key nuclides, while ⁵⁵Fe, ⁵⁹Ni, and ⁶³Ni were selected as DTM nuclides. Pearson correlation analysis revealed strong correlations between key nuclides and DTM nuclides, particularly ⁵⁵Fe/⁵⁶Co and ⁵⁹Ni/⁵⁸Co, with correlation coefficients exceeding 0.9. Further studies will be conducted to evaluate the sensitivity analysis of scaling factors, considering variables such as operating history.

Acknowledgements

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Keywords: PET cyclotron, Activation, Scaling factor method, Correlation analysis, PHITS-DCHAIN

Influence of Heavy Metals Ions on Radionuclide Adsorption in Engineered Barriers of Radioactive Waste Disposal

Yulim Lee and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

In the disposal of radioactive waste, ensuring the containment and isolation of radionuclides is a critical concern for the long-term safety of the environment and human health. Radioactive waste disposal facilities depend on engineered and natural barriers, such as concrete and soil, to prevent radionuclide migration. However, the presence of heavy metals in mixed waste could have influences in this process. This study examines the impact of heavy metals, leached from mixed waste, on the adsorption of radionuclides like Cs-137, Sr-90, and Co-60 onto barrier material.

Laboratory experiments involve exposing barrier materials including concrete and soil to groundwater solutions containing both heavy metal ions (Pb and Cd ions) and radionuclides (Cs, Sr and Co). The adsorption capacities of each barrier are measured and compared under different conditions to assess competitive adsorption.

The study enables the identification of the preferential adsorption between heavy metals and radionuclides, as well as the evaluation of how the adsorption of heavy metals affects the adsorption behavior of radionuclides. These results may guide the development of improved disposal strategies concerned with mixed waste, ensuring more effective long-term containment of radioactive and mixed waste.

Management of Sealed Disused Radioactive Sources in Nigeria

Monica Stephen Faruna* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*monicastephen6453@gmail.com

In Nigeria, radioactive materials are vital to many industries, including agriculture, education, hydrology, oil and gas exploration, and medical and industrial uses. These materials, however, become disused sealed radioactive sources when they are no longer in use and need to be handled carefully. The environment and public health can be negatively impacted by improper handling of sources, which emphasizes the significance of implementing efficient management techniques.

Nigeria Atomic Energy Commission and the Nigeria Nuclear Regulatory Authority are responsible for creating guidelines to manage disused sealed radioactive sources. These guidelines cover how to store, transport, and dispose of these sources. This is done to ensure nuclear safety and security, following international standards set by organizations like the International Atomic Energy Agency. Managing disused sealed radioactive sources is very important in Nigeria's nuclear safety plan. Even though they're no longer used, these sources still have radioactive materials inside.

This study focuses on the current state of Disused Sealed Radioactive Sources management in Nigeria. It does not delve into the technical aspects of radioactive decay or the specific uses of radioactive materials. Instead, it examines the existing policies, infrastructure, and practices for handling these disused sealed sources.

In Nigeria, disused sources are stored in a 200-liter concrete storage drum, which is used to store sources such as Sr-90, Co-60, Cs-137, Ra-226, Fe-55, Ir-192, Am-241, Cf-252, and Pu-239. The preparation of the 200-liter storage drum involves a specific process and composition. At a height of 10 cm from the bottom of the drum, the volume of the concrete is calculated as 0.0255 m^3 . A specific ratio of cement, sand, gravel, and water is used to create a concrete mixture, which is then cast inside the drum with four cross-iron rods for reinforcement. A pipe with a height of 59 cm and a diameter of 57 cm is inserted into the drum. The total volume of the concrete is calculated to be 0.146 m^3 . A second mixture is prepared according to this ratio and cast, with the pipe placed at the center to hold the DSRS capsules. The final concrete mixture has a compressive strength of 8 MPa.

To guarantee safety and sustainability, the goal of this research is to improve Nigeria's disused sealed radioactive source management system by conducting a thorough evaluation of its existing situation.

Acknowledgements

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Keywords: Disused Sealed Radioactive Sources (DSRS), Nigeria, Radioactive waste management, Nuclear safety, Environmental impact, Public health

Development of a Mechanochemical Approach for the Room-Temperature Disposal of Materials Containing Radioactive Asbestos

Kun-kyu Choi, Hyung-kwon Jeong*, Dong-gyu Jang, Jeong-won Park, Myeong-ho Kim, and Min-su Kim

Saebitene Co., Ltd.

*jhk0502@saebitene.co.kr

Asbestos is a highly hazardous substance that can cause cancer when inhaled through the respiratory system, leading to its global phase-out. In South Korea, since 2011, any activity involving the handling or movement of materials containing more than 1% asbestos has been prohibited by law. Asbestos waste from nuclear power plants is used in structural materials for building interiors, major facilities such as storage tanks, vessels, pipes, ducts, and insulation for buildings and equipment. In this study, slate, an asbestos-containing material, along with two types of insulation materials (fiberglass and gypsum) from nuclear power plants, were selected as test materials. By applying mechanochemical grinding technology using a ball mill, the crystal structure of asbestos found in nuclear power plants was altered, removing its non-acidic characteristics and effectively eliminating its toxicity. The toxicity of asbestos was dramatically removed without any separate chemical treatment at room temperature. Performance verification was conducted using phase-contrast optical microscopy (PCM), scanning electron microscopy (SEM), and X-ray diffractometry (XRD) to confirm structural and morphological changes of the constituent elements. The meaning of the material change before and after treatment is that the asbestos-containing material changed into calcium carbonate, quartz, and amorphous material through the treatment process. It was confirmed that asbestos in the slate (which contains 15% asbestos) was removed, as the needle-like structure disappeared when analyzed using PLM and SEM. However, asbestos analysis of the insulation materials from the nuclear power plant, which were initially presumed to contain asbestos, revealed no asbestos components. The experimental results demonstrated that the crystalline asbestos transformed into an amorphous phase after just a few minutes of ball milling. To industrialize this process and achieve optimal efficiency, further experiments are needed on factors such as the material and size of the ball, the oxidation degree of the cutter, the amount of the sample, and the hardness and compressive strength of the mixture. Consequently, a reevaluation of the total amount of asbestos-containing waste in the nuclear power plant, based on updated asbestos analysis, is necessary. When asbestos-containing materials are treated/disposed of at nuclear power plants, the asbestos that has been detoxified can be treated as general waste, and the same method can also be used when treating general domestic asbestos-containing materials.

Keywords: Asbestos, Mechanochemical, Ball mill, Amorphous phase

Performance Tests on Polymer Waste Form of Boric Acid Granules and Spent Ion Exchange Resins

Se Yup Lee* and You Ha Lee

Korea Nuclear Engineering Co., Ltd.

*edgelee@naver.com

Two kinds of wastes generated from domestic nuclear power plants were solidified with polymer to produce the stable waste form. One was boric acid granules as a low- and medium- level wastes from pressurized light water reactors (PWRs). The other one was spent ion exchange resins (spent resins) from CANDU reactor. The polymer which consists of polyolefin, amine compounds and antimony trioxide was used to solidify the wastes. The polymer was directly injected into a mold which is packed with the boric acid granules or the spent resins. The waste form was removed from the mold after the polymer mixture was entirely cured.

A series of performance tests specified in NRC Waste Form Technical Position (Rev.1) was conducted including compressive strength test, water immersion test, leaching test, thermal stability test, and irradiation stability test for the polymer waste forms.

Every polymer waste form containing the boric acid granules and the spent resins had exhibited excellent structural integrity of more than 34.5 MPa (5,000 psi) of compressive strength. On thermal stability testing, and water immersion testing, no degradation was observed in the waste forms. Also, by measuring the compressive strength after these tests, it was confirmed that the structural integrity was still retained.

A leach test was performed in accordance with ANSI/ANS 16.1 by using non radioactive cobalt, cesium and strontium. The demineralized water and the sea water were, respectively, used as leachant for the leach test. When the demineralized water was used as the leachant, the leaching indices of cobalt, cesium and strontium were more than 9 for the waste forms containing boric acid granules and more than 13 for the waste forms containing spent resins, respectively. When the leachant was sea water, in case of waste form containing boric acid granules, the leaching indices were similar to those when the leachant was demineralized water. On the other hand, in case of waste form containing spent ion exchange resins, the leaching indices of cobalt, cesium and strontium decreased to about 8, respectively.

Irradiation stability test was performed at the dose of 10^9 rad. No visible evidence of significant degradation was observed and compressive strengths of the waste forms were more than 500 psi.

After the performance tests for the polymer waste forms, it is believed that the polymer waste form is very stable and could satisfy the acceptance criteria for disposal.

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Keywords: Solidification, Waste forms, Polymer, Boric acid granules, Spent ion exchange resins, Spent resins

Preliminary Estimation of Radionuclide Inventory for 3rd Trench Disposal Facility

Kangil Jung*, Yuhwa Son, and Jaechul Ha
Korea Radioactive Waste Agency (KORAD)

*kangile@korad.or.kr

To accommodate the large volumes of very low-level radioactive waste (VLLW) generated during nuclear power plant decommissioning, the corporation is promoting the construction of a 3rd Trench disposal facility. Considering the total expected amount of VLLW, the plan is to sequentially expand the facility to dispose of 80% of the waste, with Stage 3-1 accommodating 160,000 units (based on 200 L per unit) and Stage 3-2 accommodating 100,000 units (based on 200 L per unit). This study focuses solely on Stage 3-1.

Accurately estimating the radionuclide inventory that the Trench disposal facility can accommodate is crucial for the smooth construction and operation of the facility.

To estimate the radionuclide inventory of the Trench disposal facility, the waste considered includes NPP operational and other operational waste (from KAERI, KNF, RI), as well as NPP decommissioning waste. Considering the long-term plan of waste, 160,000 units (based on 200 L per unit) were selected, consisting of 154,363 units of NPP decommissioning waste and 5,637 units of NPP and other operational waste.

Although the waste may arise in various forms, significant uncertainty exists regarding its final form. Therefore, estimations were made using available data, assuming radioactivity levels consistent with DAW waste. The radioactivity per package of operational and other operational waste was calculated based on actual waste accepted at the 1st silo disposal facility.

The decommissioning waste, which constitutes the largest volume, was determined based on the radionuclide inventory data provided by KHNP for the decommissioning waste from the Kori 1. While 200 L containers were applied for the disposal of operational waste, P2B, P3, and P4A (Soft bag) containers were considered for packaging the decommissioning waste. One of the three trenches was designated solely for decommissioning waste, while the other two trenches were designated for operational and other operational waste.

Based on the available data, the estimated radionuclide inventory for the 3rd Trench disposal facility was calculated to be $2.74\text{E}+14$ Bq ($42,080\text{ m}^3$, 160,000 units based on 200 L per unit).

However, the decommissioning waste data currently used is derived from numerous assumptions and contains significant uncertainties regarding radioactivity, final waste form, and characteristics.

Furthermore, there is a strong possibility that the waste and radionuclide inventory to be disposed of could change following the safety assessment of the disposal facility.

Therefore, it is essential for the waste-generating organizations and disposal operators to engage in ongoing discussion to ensure the safe and sustainable operation of the disposal facility.

Keywords: 3rd trench disposal facility, Inventory, Disposal volume, Radioactivity

Preliminary Prediction of Difficult-to-Measure (DTM) Radionuclide Concentrations in Nuclear Waste Using Machine Learning Method

Do-Hyeon Kim and Jun-Yeop Lee*

Pusan National University (PNU)

*jylee@pusan.ac.kr

Due to the continued accumulation of nuclear wastes from various industries, effective isolation methods are urgently required. However, the final disposal of nuclear waste needs detailed information on the characteristics and quantities of radionuclides contained in the waste package. Since many of these radionuclides are not easily and directly measurable, the scaling factor (SF) method, an indirect approach, has been employed for the safe management of nuclear waste.

The SF method estimates the amounts of difficult-to-measure (DTM) radionuclides, such as beta- and alpha-emitters, based on the quantity of easy-to-measure (ETM) radionuclides, typically gamma-emitters. Statistically, deriving these SFs has been traditionally dependent on the empirical models, often limited to simple parametric methods. To overcome these limitations, this study employs the XGBoost algorithms, a machine learning technique known for its high performance in regression tasks and in handling imbalanced data, to predict the concentrations of DTM radionuclides based on ETM radionuclides. Radionuclide inventory of nuclear waste, which includes the quantitative information of both DTM and ETM radionuclides, was used to construct the prediction model. Input data consisted of plant number, ^{60}Co , ^{137}Cs , and ^{144}Ce , while output data included ^3H , ^{14}C , ^{55}Fe , ^{63}Ni , ^{90}Sr , ^{99}Tc , ^{129}I , $^{238+239+241}\text{Pu}$, ^{241}Am , and $^{242+244}\text{Cm}$. The performance of prediction model derived in this study was evaluated based on R^2 and RMSE values.

As a result, the prediction model demonstrated reasonable performances for ^{63}Ni , ^{55}Fe , ^{90}Sr , and actinides, with R^2 and RMSE values ranging from 0.7720 to 0.8866 and from 0.7317 to 0.8622, respectively. In contrast, the prediction results for ^3H , ^{14}C , ^{99}Tc , and ^{129}I were relatively less successful, with R^2 values between 0.3583 to 0.5776 and RMSE values from 0.8566 to 1.0949. Among the DTM radionuclides, ^{63}Ni showed the highest performance, while ^3H exhibited the lowest performance. The results of this study are seemingly comparable with the tendencies of radionuclides determined with the Pearson's Correlation Coefficient (PCC) method.

The results of this study suggest that the application of machine learning can be a novel approach to addressing challenges in nuclear waste management and can also be used to cross-validate the concentrations of DTM radionuclides determined with other classical methods. Nevertheless, follow-up researches focusing on enhancing the prediction performances by employing the expanded input database and in-depth hyperparameter tuning are assessed to be necessary. Furthermore, the complementary analyses of the importance of input variables are also expected to increase the performance of the prediction model.

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Keywords: Scaling factor, DTM, ETM, Machine learning, XGBoost

Development and Demonstration of Technology for Ensuring Disposal Suitability of Spent Resin in Pressurized Heavy Water Reactor

Hyeon Oh Park¹, Kyu Ho Lee^{1,*}, Dong Hyun Kim¹, Hee Jin Kim¹, Ki Hyun Kwon¹, Seung Geon An¹,

Ha Young Oh¹, Hwan Seo Park², and Ki Rak Lee²

¹Sunkwang T&S Co., Ltd.

²Korea Atomic Energy Research Institute (KAERI)

*khlee@sktns.co.kr

With the permanent shutdown of Wolseong Unit 1, Pressurized Heavy Water Reactor (PHWR) in Korea, the Nuclear Power Plant (NPP) decommissioning project is in full swing, and interest in the development of decommissioning technology for unproven PHWR is increasing. The spent resin in the spent resin storage tank of the PHWR has a high radioactivity concentration of C-14 nuclide, which is classified as Intermediate Level Waste (ILW), so it is necessary to develop technology to treat the spent resin and ensure its suitability for disposal.

Therefore, in this study, we have developed a 'PHWR mixed waste treatment technology' that can treat spent resin stored in the spent resin storage tank of PHWR. The 'PHWR mixed waste treatment technology' consists of a 'mixed waste separation technology' that separates the mixed waste stored in the spent resin storage tank into spent resin and mixture (activated carbon and zeolite), a 'spent resin treatment technology' that removes C-14 nuclides from the separated spent resin, and a 'capture technology' that captures the removed C-14 nuclides. To verify the performance of the above three technologies, a large-scale 1-ton facility was built and a demonstration experiment was conducted at a domestic PHWR.

Demonstration experiment confirmed the separation of spent resin and mixture in the mixed waste, and the specific radioactivity of the C-14 nuclide was analyzed through the separated spent resin, which reaffirmed that the spent resin was considered to be ILW. The separated spent resin was subjected to 'spent resin treatment technology', and it was confirmed that the radioactivity concentration of the spent resin was reduced from Intermediate to low levels, thereby verifying 'Ensuring the suitability of spent resin for disposal'. The C-14 nuclide removed from the spent resin was also captured, and it was confirmed that the concentration of C-14 nuclide in the facility was reduced to below the atmospheric level. In addition, the soundness and safety of the facility was verified as there was no leakage of mixed waste or leakage of C-14 nuclides despite being a large-scale 1-ton facility.

Therefore, through the demonstration experiment of a large-scale 1-ton facility using the technology developed through this research, it was confirmed that the technology can be applied to PHWR, and it is expected to contribute to the development of the NPP decommissioning industry in Korea or abroad.

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Keywords: PHWR, Spent resin, C-14, Disposal safety, Decommissioning

Thermodynamic Modeling of Hydrate Assemblages and Porewater Composition in Blended Cements With Supplementary Cementitious Materials

Dayeong Lim¹, Sol-Chan Han², and Yongheum Jo^{1,*}

¹Hanyang University

²Lawrence Berkeley Natl. Lab.

*yongheumjo@hanyang.ac.kr

Engineered barriers of a radioactive waste repository play a critical role in preventing water intrusion into the facilities and retarding radionuclide migration. Cementitious materials such as concrete have been widely used as engineered barriers in low- and intermediate-level radioactive waste repositories due to its benefits. For instance, cement hydrates formed by the hydration of cement clinkers can enhance the structural stability of the repository. In addition, different types of cement hydrates could provide various surface functional groups which can adsorb radionuclides, so that minimize the release of radionuclides to the surrounding environment. Several repository design concepts are considering blending ordinary Portland cement (OPC) with supplementary cementitious materials (SCMs) in order to reduce construction costs and carbon footprint. Since the chemical composition of SCMs is greatly different from that of OPC, the blending of SCMs with OPC could result in the alteration of hydrate assemblages and geochemical conditions of cement porewater compared to neat OPC hydrates. The alteration of such chemical characteristics of blended cementitious barriers, eventually, may affect the safety function of the barriers. In this context, it is important to understanding the effect of blending SCMs with OPC on cement hydrates assemblages.

This study aimed to investigate the changes in the formation of cement hydrates with varying SCM compositions in a blended SCM-OPC system. In particular, fly ash, blast furnace slag, and silica fume were SCMs of interest. The hydrate assemblages of blended cements were calculated by means of thermodynamic modeling using geochemical simulation code (PHREEQC) and a thermodynamic database for cement system (CEMDATA 18). The simulation results showed that the addition of SCMs significantly changes the types of formed cement hydrates and their abundance in the hydrated cement system. Accordingly, the chemical characteristics of porewater also differed depending on the blending ratio. The simulation results were compared with experimental data from various literature and the expected effects caused by changes in cement hydrates on the safety function were discussed.

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Keywords: Cement hydration, Supplementary cementitious materials, Geochemical modeling

Update on Treatment of Large Concrete Drum for Safe Management and Disposal

Young Hwan Hwang*, Seong-Sik Shin, Sunghoon Hong, Seok-Ju Hwang, Jung-Kwon Son, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

The large concrete drums, which contain boron concentrate cement solidified form and spent resin, need to be properly treated and sent to disposal facility for final decommissioning of Kori unit 1. Since the inner drums are surrounded by concrete drums and grouting concrete, the segmentation process and treatment processes for satisfying waste acceptance criteria (WAC) are required. In this paper, the development of equipment and of pilot test results will be discussed. The mockup pilot test is implemented to verify the treatment technology.

Keywords: Large concrete drums, Waste acceptance criteria, Historic waste, Cement solidification

4분과

중·저준위폐기물관리 (POSTER)

Adsorption Performance of Petroleum Pitch-based Activated Carbon for Radionuclides: Exploring Alternatives for Nuclear Power Plant Liquid Waste Treatment

Hyeongmin Park*, Heungnae Lee, Guihwan Jo, Yongsoo Ahn, Myungpil Kim, and Junhwi Bang

Korea Standards Analysis Corp.

*hyeongmin.park@ko-sa.co.kr

Currently, activated carbon used to treat liquid waste from nuclear power plants is entirely dependent on imports, which causes resource dependency problems such as the shortage of urea in diesel vehicles. This study evaluates the adsorption performance of extruded and spherical activated carbon produced from domestic petroleum residues, aiming to verify its potential as an alternative to existing activated carbons.

Experiments utilizing the Jar Tester apparatus assessed the adsorption rates for radionuclides Co-60, Cs-137, and I-129. The results showed that petroleum pitch-based activated carbons, both bentonite-mixed and phosphate-treated, exhibited superior adsorption performance compared to conventional coconut-based activated carbon. Specifically, bentonite-mixed granular and spherical activated carbons demonstrated adsorption rates of 34% (Co-60), 67% (Cs-137), 57% (I-129) and 18% (Co-60), 39% (Cs-137), 66% (I-129), respectively. These findings indicate that petroleum pitch-based activated carbons have significant potential as viable substitutes in the treatment systems of nuclear power plant liquid wastes.

Acknowledgements

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Keywords: Petroleum pitch, Activated carbon, Radionuclides, Absorption performance

Development of Purge Gas Recombiner System Based on Catalytic Recombiners and Adsorbers for Tritium Removal

Soon-hwan Son*, Houn-gon Cha, and Ji-seon Kim

AGENCORE Co.

*shson@agencore.net

AGENCORE is developing and installing various tritium handling systems to produce the gaseous tritium light sources (GTLS). The tritium handling systems should be installed in a high-integrity glove box in preparation for tritium leakage. If tritium leaks from components containing tritium, a provision to remove elemental tritium leaked into the glove box is required to minimize the release of tritium into the environment. AGENCORE has designed and installed the new purge gas recombiter system (PGRS) using both catalytic recombiners of Ni metal-based reactor, and molecular sieve (MS) adsorbers for the tritium handling glove boxes. The PGRS includes a surge tank, a recirculation pump, a heat exchanger, catalytic recombiners (2 parallels), gas preheaters, MS adsorbers (2 parallels), a hygrometer, and tritium monitors. When the tritium concentration in the glove box exceeds the set level (5.0×10^8 Bq/m³), a tritium high alarm is generated, all flow paths outside the glove box are blocked, and the PGRS is connected to start the recirculating operation. The glove box atmospheric gas passes through the gas preheater in advance, the temperature rises to about 250°C, and then enters the catalytic oxidizer. More than of 90% of the elemental tritium introduced into the catalytic oxidizer is converted into moisture by reaction with oxygen. The gas that comes out of the catalytic recombiter is cooled to below 25°C while passing through the heat exchanger and enters the MS adsorber. In the MS adsorber, moisture is completely removed at a dew point temperature of -20°C or less, and only gaseous stream is exhausted. The gas that comes out of the MS dryer passes through the tritium monitor again. If the tritium concentration level is below the set value (3.7×10^6 Bq/m³), it may be discharged by the exhaust fan, but if it is above the set value, it should be recycled to the glove box and processed. This system will be used to remove tritium in the glove box including the GTLS production equipment of AGENCORE. Currently, the license for the GTLS manufacturing facilities with the PGRS system is being reviewed.

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Keywords: Tritium, Purge gas recombiter system, Catalytic recombiter, MS adsorber

Evaluation of Physicochemical Properties of TEDA-Impregnated Activated Carbon for Heating, Ventilating, and Air Conditioning (HVAC) System in Heavy Water Reactors

SooHwan Kim¹, JunHeon Lee¹, GwanHyun Lee², and Jun-Hyung Ryu^{1,*}

¹Dongguk University

²Hi-Air Korea Co., Ltd.

*jhryu@dongguk.ac.kr

The physicochemical properties of TEDA (Triethylenediamine)-impregnated activated carbon, which is used in Heating, Ventilating, and Air Conditioning (HVAC) system of a heavy water reactor and adsorbs various types of gases through van der Waals forces with porous carbon, were analyzed. TEDA-impregnated activated carbon is widely used to remove radioactive iodine and methyl iodine in heavy water reactors, and its performance and stability are important factors for the safe operation of Pressurized Heavy Water Reactor (PHWR). In this study, the surface properties, pore structure, thermal stability, and chemical bonding state of TEDA activated carbon were evaluated using various analytical techniques. The microstructure and thermal/chemical properties of TEDA activated carbon were quantitatively analyzed using scanning electron microscopy (SEM), nitrogen adsorption-desorption experiment, thermogravimetric analysis (TGA), and X-ray photoelectron spectroscopy (XPS). As a result, it was confirmed that TEDA activated carbon has a high specific surface area and microporous structure, high thermal stability, and strong TEDA bonds are formed on the surface. These characteristics suggest that TEDA-impregnated activated carbon is suitable for efficient removal of radioactive iodine and long-term use in the air purification system of heavy water reactors.

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Keywords: Radioactive carbon, Physicochemical properties, TEDA

Condensate Reduction Strategies in Radiation-Controlled Areas Using Outdoor Air Conditioning (OAC) Systems

Yongsang Lee*, Youngwoo Kwon, and Seonho Noh

Korea Atomic Energy Research Institute (KAERI)

*ys2@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI) contains numerous radiation-controlled areas. One such area, the Nuclear Cycle Experiment Facility, operates 15 air conditioning units during the summer to maintain the temperature and humidity of experimental equipment.

The condensate generated by these air conditioners exceeds 10 tons annually, which must be treated as low-level radioactive waste, consuming significant time and financial resources.

This study aims to utilize Outdoor Air Conditioning (OAC) systems in radiation-controlled areas to maintain temperature and humidity for equipment, significantly reduce condensate production, and thereby reduce the associated time and costs.

Keywords: Condensate reduction, Outdoor air conditioning, Radiation-controlled areas, Environmental safety, Radioactive contamination management

Plan for Building a Standardization and Implementation Platform for Solid Waste Analysis and Evaluation

Hee-Jin Ahn¹, Dae-Seok Hong², and Seung-Chul Park^{1,*}

¹Green Radiation Co., Ltd.

²Korea Atomic Energy Research Institute (KAERI)

*scpark@greenrad.kr

In nuclear power plants and industrial institutions, solidification or fixation treatment is essential for various difficult-to-dispose radioactive wastes. In order to deliver the solidified waste to a disposal site, structural safety (compressive strength test, immersion test, thermal cycle test, radiation exposure test), leachability (leach test), and free water measurement are to be performed. A lot of time and money are being spent on the process of analyzing and evaluating whether the structural safety of the solidified wastes has been secured, but detailed test procedures and standardized evaluation methods suitable for the domestic environment have not been established. As of now, only a few drums of cement-fixed wastes have been disposed of in domestic disposal sites, and there is a lack of preparation for the timely disposal of large quantities of wastes in the future. Currently, the evaluation of the characteristics of the solidified wastes is being conducted in a non-standardized manner, making it difficult to secure certified reliability for the entire evaluation procedure. Therefore, in order to timely dispose of large quantities of solidified wastes in the future, reliability improvement through advanced analysis and evaluation systems is necessary.

Keywords: Solid waste form, Analyzing and evaluating, Material characteristics, Structural stability, Disposal

Development of Radioactive Waste Activated Carbon Treatment Technology Applying Mechanochemical Technology Based on Room Temperature and Eco-friendly Methods

Kyeong-Ryeong Kim^{1,*}, Sang-Soo Kwak¹, Yong-Ho Hong¹, Dong-Sik Jin¹, and Byoung-Sub Han^{2,*}

¹Nuclear Environ. Technology Co.

²Korea Advanced Saf. and Protection Inst.

*redmonac@netec.co.kr, bdothan@gmail.com

Research on self-disposal through treatment of radioactive waste activated carbon is being conducted domestically and internationally. However, there is no technology to separate radioactive carbon dioxide ($^{14}\text{CO}_2$) from radioactive waste activated carbon in order to satisfy the permissible concentration criteria of self-disposal. This study is about the treatment methods and treatment equipments for self-disposal through treatment of radioactive waste activated carbon, and is the technology that can be reduced the radioactivity to the permissible concentration of self-disposal. The radioactive waste activated carbon treatment technology using ball mill equipment can be prevented the dispersing of fine activated carbon in the pulverization process using ball mill (mechanical) and wet methods (chemical), has low explosion risk, can be treated at room temperature, an eco-friendly and safe. In addition, it is a mechanochemical treatment methods that does not generate additional CO_2 or secondary waste through the carbon nonactivation process during the treatment process, and has the advantages of being compact and economical due to the simple process and equipments configuration.

Stage 1 of the treatment technology is pulverization and desorption process, radioactive waste activated carbon is injected into a ball mill equipment and pulverized into nanometer sizes by interacting the impact and abrasion principles. In this study, a reagent was used to desorb CO_2 adsorbed in micropores of radioactive waste activated carbon, and usable solvents include NaOH , KOH and LiOH , etc. When reagent solution is injected, carbonate and H_2O are generated, and the pulverization and desorption process is performed for approximately 4 hours. In stage 2 recovery process, the product generated in stage 1 is transferred to a recovery tank, and when calcium hydroxide is injected into the recovery tank, carbonate is precipitated. The precipitated carbonate is treated as radioactive waste, and the supernatant, waste activated carbon, is classified as a self-disposal waste. Radioactive waste activated carbon samples, which were not reduced after repeated treatment with other treatment technologies to treat radioactive waste activated carbon, were subjected to treatment process to pulverization, desorption, and recovery applying ball mill technology, and then radioactivity was analyzed. The results of the radioactivity analysis confirmed that the radioactivity could be reduced to the level of 1.0 Bq/g , which is the permissible concentration criteria of self-disposal for ^{14}C .

When the radioactive waste activated carbon is treated applying the mechanochemical process presented in this study, radioactive carbon (^{14}C) can be sufficiently removed to permissible concentration of self-disposal. Radioactive waste activated carbon treatment technology is expected to reduce the cost of managing radioactive waste by reducing the generation of radioactive waste and contributed to the effective space utilization of disposal facility, and the same technology can be applied to the treatment of radioactive spent resins. The results of this study can be used as basic data of treatment technology for radioactive waste activated carbon, and following research and optimizing process is needed to satisfy the permissible concentration of self-disposal of ^{14}C in radioactive waste activated carbon.

Keywords: Mechanical chemical process, Room temperature treatment, Eco-friendly technology, Carbon nonactivation process

Verification of Heat Treatment Process Equipment for Reducing DAWs

Jihyun Yu, Seonghun Kim, Hyerim Kim, Jangsoo Suh, and Younjung Choi*

Sae-An Enertech Corp.

*yjchoi@sae-tec.com

A large amount of DAWs (Decommissioning Activities Waste) is generated from the operation and maintenance of nuclear facilities, and volume reduction of DAWs is a critical issue due to the limited space available for on-site storage and geological disposal of waste.

Thermal treatments such as incineration can significantly reduce the volume of waste, but they also generate CO₂, a greenhouse gas, during the thermal decomposition of waste, as well as pose risks of emitting acid gases and dioxins. Therefore, a thermal treatment process that minimizes the generation of CO₂ and harmful gases is needed.

In this study, we conducted radiological and physicochemical analyses of combustible waste to develop an incineration-alternative process that achieves high volume reduction efficiency similar to incineration, while minimizing the emission of CO₂ and harmful gases. Based on the results of these analyses, we designed the necessary equipment.

The incineration-alternative process is carried out in three stages, with the conditions for each stage as follows: Stage 1. Sealed low-temperature thermal treatment using temperature and pressure (~250°C), Stage 2. High-temperature thermal treatment under an inert atmosphere (900~1,000°C), Stage 3. High-temperature chlorination (900~1,000°C) for the removal of Cs and Co.

To evaluate the laboratory performance of the process, we conducted assessments of each unit using decontamination paper. Since the high-temperature thermal treatment and high-temperature chlorination are performed using the same equipment, only the high-temperature thermal treatment process was carried out, and it was determined that there were no issues with the equipment's performance evaluation.

After thermal treatment, the volume of the decontamination paper was reduced to approximately one-fifth of its original size. Elemental analysis (EA) revealed that the carbon retention rate was 65.9% after the sealed low-temperature thermal treatment, 83.03% after the high-temperature thermal treatment, and 86.7% when the sealed low-temperature and high-temperature treatments were combined. This indicates that the introduction of sealed thermal treatment can increase the mass retention rate (equivalent to carbon retention rate) and minimize gas generation during the process.

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Keywords: Dry active waste, Alternative incineration, Thermal decomposition, Volume reduction

Comparison of Leaching Behavior of the Geopolymer Waste Forms With Cs, Sr, Co, Re, and I

Seongye Kwon and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Korea is currently operating a low- and Intermediate level radioactive waste disposal site at Gyeongju. In order to dispose radioactive waste in a disposal site, the acceptance criteria must be satisfied. At this time, the acceptance criteria of leachability index directly related to leakage of radionuclides based on leaching behavior of radionuclides. The Gyeongju disposal site should meet a leachability index of 6 or higher by performing a leaching test with the ANS 16.1 test method.

ANS 16.1 is the leaching test method using Deionized water as leachant. However, the leachant in the actual disposal site is expected to have the high pH condition due to the reaction of concrete structures and groundwater. Therefore, the leaching test method based on the actual disposal environment is required. For considering the actual environment, the leaching test must use Alkali-state water to reflect the scenario of reaction with concrete and groundwater.

For disposal of low- and Intermediate- level radioactive wastes, the Geopolymer have been widely studied to immobilize anionic and cationic species. However, few studies have focused on the stabilization of anionic radionuclides. Cs, Co and Sr were represented the main radionuclides of cationic species. Re and I were tested for anionic radionuclides.

In this study, the immobilization of geopolymer waste with anionic species such as Re and I depend on three types of concentration (0.1, 1, 10 mmol). Also, cationic species with Cs, Sr, and Co was tested according to the ANS 16.1 method. For the leaching test, Leachant was used two types of Deionized water (DI) and Concrete-saturated Ground water (CGW) for considering of disposal environment.

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Keywords: Leachant, Leaching behavior, Leachability index, Radioactive waste, Repository

Evaluation of Stability and Performance of Solidification Materials for Safe Disposal of Radioactive Waste: A Review

Sehyun Cho and Jaehyuk Kang*

Jeju National University (JNU)

*jaehyukkang@jejunu.ac.kr

The safe disposal of radioactive waste is a critical challenge in contemporary nuclear power generation. The radioactive isotopes of primary concern include cesium, strontium, cobalt, and iodine. The release of these isotopes into the environment can have severe impacts on human health and the ecosystem. As a result, extensive research has been conducted on various materials to safely solidify waste containing these radioactive isotopes. Representative solidification materials include: 1) Ordinary Portland Cement, which forms a calcium-silicate-hydrate structure through a hydration reaction; 2) Alkali-Activated Cement, which uses an alkali activator to enhance binding properties; and 3) Geopolymer, which utilizes an alkali activator to create a gel network structure.

To understand the solidification mechanisms of binding materials and their advantages and disadvantages, numerous studies have been conducted. Therefore, it is essential to review and summarize the studies related to solidification mechanisms and characteristics of each material. Thus, the main objective of this study is to review the diverse materials used for the solidification of radioactive waste, focusing on their solidification mechanisms and evaluating their performance based on waste acceptance criteria test conducted in various countries.

This review contributes to future radioactive waste management strategies by identifying the most suitable materials for safe disposal. It also addresses gaps in current solidification research and provides insights for future investigations.

Keywords: Solidification, Geopolymer, Ordinary portland cement, Alkali-activated cement, Radioactive waste

A Study on Classification for Treatment of Non-Fuel Materials in the SFP of PWR

Kyungho Roh*, Beomgyu Kim, Taehyung Na, and Donghee Lee

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*rohkyungho@khnp.co.kr

Domestic PWR (Pressurized Water Reactor) consists of 22 operating nuclear power plants, one permanently shutdown reactor which is a WH type plant, and one reactor awaiting continued operation. Inside the spent fuel pools (SFP) of each PWR, various non-fuel materials exist in addition to used nuclear fuel. Among these materials, we have listed those stored in the SFPs of the permanently shutdown reactor and the oldest one OPR (Optimized Power Reactor) that has been in operation for the longest period. In the SFP of the permanently shutdown reactor, there are skeleton of spent fuel assembly, baskets including in-core instrument, baskets including damaged fuel rod, baskets including, baskets including neutron source assembly, baskets including reactor structure component, baskets including irradiated specimen, control rods, poison rods, thimble plug, etc. inserted on the spent fuel, while in the SFP of the oldest OPR, there are baskets including control rods, baskets including in-core instrumentation, dummy fuels, baskets including damaged fuel rods, baskets including waste generated after fuel repair, filter and pump resulting from the equipment when ultrasonic cleaning of fuel. Due to the variety of non-fuel materials being stored, different methods must be considered for their treatment and disposal. Additionally, considering the design capacity of domestic intermediate and low-level disposal facilities, it may become impossible to dispose of intermediate level non-fuel materials. In this regard, in this paper, we classified non-fuel materials stored in spent fuel pools of two nuclear power plants mentioned earlier as items that should be temporarily stored and disposal methods established later based on preliminary activation assessments and engineering judgments, and those that can be disposed of after suitability for disposal is reviewed. Items deemed suitable for disposal will be prioritized and nuclide analysis will begin to ensure suitability for disposal. Future plans include developing storage solutions for temporarily stored items, including handling tools, and conducting research of necessary technologies for radionuclide identification during nuclide analysis as a part of disposal suitability review.

Keywords: Non-fuel material, Temporary storage, Nuclide analysis, Disposal suitability, Intermediate level

Role of Getters in Preventing Tc Leaching in Radioactive Waste Disposal: A Review

Hyolee Kim and Jaehyuk Kang*

Jeju National University (JNU)

*jaehyukkang@jejunu.ac.kr

Technetium (Tc), one of the fission products, primarily exists in the Tc(VII) species as pertechnetate (TcO_4^-). Due to its long half-life (2.13×10^5 years) and high mobility, it can easily move through the environment when it is disposed of in waste form. If Tc is released, it can enter the food chain and eventually be ingested by humans, potentially causing harmful effects. While reduction of Tc to the Tc (IV: TcO_2) species can lower its mobility, it is unstable and prone to rapid reoxidation, which highlights the need for stable treatment and disposal methods.

Waste containing Tc is currently classified as low and intermediate-level radioactive waste and should be treated and disposed of in low-temperature waste forms such as cement. However, the cementitious materials have a negative structural charge, and since TcO_4^- also exhibits the same charge, it is difficult to immobilize it within the cement. This can cause Tc to be released into the environment again.

Therefore, this study reviewed various approaches to prevent Tc leaching into the environment during the disposal of Tc-containing waste forms. Several studies have proposed incorporating materials that can sorb specific radionuclides and metallic pollutants into waste forms to address the leaching issue. These materials are known as “getters” and function as sorbents within the waste forms. This review examined the mechanisms and efficiency of Tc immobilization in waste forms containing getters. However, since there is limited research on the use of getters within waste forms, this review also covered sorbents that function outside of them. By assessing these multifaceted approaches, we aim to provide a comprehensive overview of current strategies and future directions in managing and containing Tc, ensuring the safe and sustainable disposal of radioactive waste. In conclusion, this review has examined various applications of getters used to prevent Tc leaching into the environment during the disposal of Tc-containing waste forms and has outlined future research directions for Tc disposal.

Keywords: Technetium, Waste form, Getter, Solidification, Immobilization

A Study on the Variation in LSC Measurement Performance With the Cocktail Composition of Analyzed Samples

Simon Oh, Kyunghun Jung*, and Jungbo Yoo

Korea Atomic Energy Research Institute (KAERI)

*kyunghunjung@kaeri.re.kr

For the permanent disposal of low- and intermediate-level radioactive waste, it is essential to quantify the specific concentrations of alpha, beta, and gamma nuclides within the waste drums. Liquid Scintillation Counters (LSC) are frequently employed to evaluate the radioactivity concentrations of beta-emitting nuclides. LSC is extensively utilized in both research environments and practical applications due to its capability to analyze a broad spectrum of nuclides and provide photon energy spectra as part of the analysis output. Consequently, the KAERI Radioactive Waste Chemical Analysis Center employs LSC to determine concentrations of H-3, C-14, Fe-55, Ni-63, and Sr-90. Nevertheless, given that LSC samples are mixed with an organic cocktail to induce photon generation, the samples transform into organic liquid waste once the analysis is concluded. Considering the complexities involved in managing radioactive organic liquid waste, it is prudent to minimize the volume produced during the research process. The generation of organic waste is contingent upon the quantity of cocktail mixed with the analytical samples, and the composition of the cocktail directly influences the efficacy of the LSC measurements. In this study, we investigated the variability in metrological performance for Fe-55 and Ni-63 as a function of the cocktail volume in metrological samples, with the objective of determining whether it is feasible to reduce the cocktail volume without compromising measurement accuracy.

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Keywords: Radioactive waste, Radionuclide inventory, Low- and intermediate-level radioactive waste, Liquid scintillation counter, Organic cocktail

Development of Cellulose Treatment Technology Using Plasma

Jeongsu Jeong* and Sunghoon Hong

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*happytiger13@khnp.co.kr

When operating a nuclear power plant, various types and types of radioactive waste are generated. Radioactive waste is a radioactive material or a substance contaminated accordingly and is subject to disposal. Due to stricter regulations on the disposal of radioactive waste, unclear acquisition standards, and lack of processing technology, large amounts of radioactive waste generated from nuclear power plants are stored for a long time. Plasma treatment technology was developed for stable comprehensive management of radioactive waste generated from nuclear power plants. Plasma processing technology is a technology that uses an electric arc phenomenon such as lightning to reduce volume with more than 1,600°C. Cellulose-containing radioactive waste (paper, cotton, etc.) generated from nuclear power plants is difficult to dispose of due to stricter regulations on the Gyeongju disposal site. Therefore, using plasma treatment technology. Cellulose waste will be treated and disposed of through empirical tests, and research will be conducted to enable actual disposal. KHNP will continue to study plasma treatment technology to ensure that radioactive waste generated from nuclear power plants can be managed stably.

Keywords: Plasma, Waste, Volume, Cellulose

Analysis of Self-Generated Colloids From Concrete and Glass Waste in Concrete Pore Water and Rainwater

Chongmyung Jin, Stuart Aberdeen, Richard I. Foster, and Sungyeol Choi*

Seoul National University (SNU)

*chosys7@snu.ac.kr

After the observation of plutonium migration due to colloids in Nevada, the impact of colloids has become an important factor in the safety assessment of radioactive waste disposal sites. Therefore, it is crucial for the impact of colloids on radionuclide migration to be determined when assessing the safety of radioactive waste disposal sites. This influence is typically evaluated through a ‘colloid ladder’. This study specifically focusing on the first step of the colloid ladder, determining whether colloids are present.

Concrete, expected to be produced during nuclear power plant decommissioning, and glass fibers, potentially produced from waste filters, were selected as particular waste materials. The experiment was designed to select particulate materials and analyze the colloids generated from these materials in leachate. The concrete, provided by Korea Radioactive Waste Agency (KORAD), was the same type used in Stage 2 of the Gyeongju Low- (LLW) and Intermediate-Level Waste (LILW) Repository was crushed to a particles size of 200 μm . Similarly, the glass fibers used in the experiment were also prepared to a particle size of 200 μm . The experiments were divided into non-immobilized and immobilized leaching tests, where immobilization was conducted using cement. The solidification process followed the KS F 2403 standard, with specimens manufactured in dimensions of 22 mm in diameter and 45 mm in height. The waste, water, and cement ratios were determined based on the KS L 5109 standard, with adjustments made to account for the potential free water generation. The waste loading was prepared in two mass ratios: 25wt% and 45wt%. The leachate used for testing was prepared in the laboratory based on the composition of rainwater collected from the Gyeongju LILW Repository and pore water from the concrete used in Stage 2 of the repository. Leaching experiments were conducted over 40 days, with samples collected and analyzed at 10-day intervals. The physical and chemical stability of the samples were characterized using Dynamic Light Scattering (DLS), meanwhile the hydrodynamic diameter, particle size distribution, and zeta potential were analyzed using electrophoresis. Additionally, the concentrations of Ca and Si were obtained from Inductively Coupled Plasma Optical Emission Spectrometry (ICP-OES).

The results showed that the concentration of Ca increased in the concrete, and the concentration of Si increased in the glass particles. Additionally, DLS analysis showed that the particle size increased over time. In the glass particle leachate, a zeta potential between -20 and -30 mV was measured, indicating the potential for colloids to exist stably. The results of this study provide crucial insight for evaluating the long-term safety of the Gyeongju LILW Repository by assessing whether colloids are self-generated from particulate waste materials.

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Keywords: Colloids, Radionuclide migration, Gyeongju LILW repository, Particulate waste, Zeta potential

A Conceptual Study on Waste Solidification Container Using Induction Melting

HyungJun Kim* and Jihyun Kim

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*junasen@kps.co.kr

The solid radioactive wastes after reduction treatment like plasma, incineration, compression and so on, then poured in a waste container with solidification phenomena over time. But, waste solidification needs to satisfy the waste package's solidification & encapsulation requirements by intermediate and low level radioactive waste acceptance criteria.

Within this paper, therefore, ways and means will be discussed to meet solidification requirements and test criteria of the solidification waste as a waste package's solidification & encapsulation requirements. And the concept of waste solidification container using induction melting will be introduced to meet solidification & encapsulation requirements and test criteria of the solidification waste for the waste package disposal.

Keywords: Solid radioactive wastes, Slags, Waste solidification, Induction melting waste package's solidification & encapsulation requirements

A Comparative Study on Plasma Arc as a Cutting Technology for Radioactive Structures

Jihyun Kim* and HyungJun Kim

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*safetyji@kps.co.kr

The internal structures of the nuclear reactor are the facilities closest to the nuclear fuel assemblies. They are called reactor internals including upper & lower internal, core barrel, thermal shield, core support plate and so on. They became highly radioactive for a long time operation of nuclear power plant, so the entire amount of radioactive structures should be cut underwater during disassembly, packaging for radioactive structures transportation also should be done underwater.

A plasma arc is a kind of energy that utilizes an electric arc to dissociate a gas and transfer an electric energy to the gas to generate very high temperature flame. This study will compare plasma arc cutting with CAMC (Contact Arc Metal Cutting) and mechanical cutting like band saw, wire saw, wheel saw underwater condition. And try to apply proper cutting technology according to disassembly restrictions underwater.

Keywords: Plasma arc, Disassembly, Cutting technology, Radioactive structures, Contact arc metal cutting, Mechanical cutting

Radioactivity Sensitivity Analysis for Bio-Shield Concrete in CANDU Reactor

Seok Geun Cho and Chang-Je Park*

Sejong University

*parkcj@sejong.ac.kr

The decommissioning of nuclear power plants, particularly CANDU reactors, presents a unique set of challenges due to the complex nature of the reactor components and the variety of radioactive wastes generated. The accurate calculation of radioactive sensitivity in bio-shield concrete ensures safe and efficient decommissioning processes. This study aims to investigate the specific activity sensitivity of bio-shield concrete in CANDU reactors by evaluating the impact of composition tolerance on the activation of radionuclides.

MCNP modeling was performed to obtain neutron flux information for major structures in the CANDU reactor. To calculate the neutron flux in the bio-shield, FMESH4 was used. The bio-shield concrete of the reactor was divided into 5 mesh sections, each 4 cm in the radial direction. In the axial direction, the reactor was divided into 100 mesh sections.

For the activation analysis according to composition tolerance, the concrete compositions of 12 NPPs listed in the NUREG-3474 report were used. The activation analysis was conducted for three cases: compositions without impurities, compositions with impurities at the middle value of the range, and compositions with impurities at the maximum value of the range. A neutron flux of $7.027\text{E}+09 \text{ \#}/\text{cm}^2$ was used. In the case without impurities, 11 NPPs were classified as very low-level waste, and 1 NPP (Susquehanna) was classified as low-level waste. However, when impurities were included at either the middle or maximum range values, an increase in the specific activity of Co-60 was observed in 11 NPPs (excluding Waterford), leading to their reclassification as low-level radioactive waste.

Additionally, the sensitivity of specific activity to neutron flux was analyzed. For this, the concrete composition of the Waterford NPP was used, with impurity values set to the median of the range. A total of 10 flux values, including the minimum and maximum flux values calculated using MCNP for the bio-shield concrete, were used. The results showed that all flux ranges ($8.27\text{E}+08 \text{ \#}/\text{cm}^2 \sim 7.027\text{E}+09 \text{ \#}/\text{cm}^2$) were classified as very low-level Waste. However, at a flux level of $5.49\text{E}+07 \text{ \#}/\text{cm}^2$, the criteria for self-disposal were met.

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Keywords: Activation analysis, Bio-shield, Concrete, Composition tolerance

Case Study on the Barnwell Disposal Facility for Domestic PE-HIC Disposal

Changgyu Kim*, Kwangsoo Park, Chungsik Lim, Haewoong Kim, Woo Nyun Choi, Taeseob Lim,

and Donghun Pak

Doosan Enerbility

*changgyu1.kim@doosan.com

The disposal plan for the KORAD phase 1 includes a disposal plan for non-standard drums including PE-HIC (Polyethylene High Integrity Container) on top of the disposal silo, so it is necessary to secure pre-disposal requirements for PE-HIC. The US NRC did not approve the use of PE-HIC alone, as it evaluated that PE-HIC did not meet the design criteria in terms of creep and bucking, radiation effects, and mechanical properties. Afterwards, there was an experience of disposing of it by overpacking it in a concrete container. In Korea, it is expected that overpacking will be necessary for the disposal of PE-HIC, and for disposal suitability review and approval by disposal operators, evaluation of long-term integrity and development of overpackaging containers suitable for the disposal environment are essential.

The use of PE-HIC is one example of a case where the NRC concluded that a stabilization product cannot meet the 10 CFR Part 61 requirements for waste form stability in the absence of special measures at the disposal site. The PE-HIC described in reports submitted by three suppliers (Chem-Nuclear, Hittman, and TFC Nuclear) were not approved by the NRC. However, the list of stabilization media and containers approved by the State of South Carolina (since 1981) confirms that certificates of conformity have been issued for several PE-HICs. PE-HIC was accommodated as an interim disposal facility at the Barnwell site while the State of South Carolina implemented special disposal regulations beginning April 1, 1989. NRC-licensed waste generators seeking to use PE-HICs for Class B and C waste disposal must either (1) transport the PE containers in an approved HIC or (2) obtain assurance and documentation from the repository operator that structural integrity consistent with 10 CFR Part 61 requirements will be provided at the repository. The State of South Carolina has authorized the operator of the Barnwell Disposal Facility to accept class B and C wastes in PE-HIC and dispose of them in concrete overpacks.

In this case study, the Operating Guidelines (FO-AD-002) and Overpack Handling Procedure (FO-OP-019) for PE-HIC manufactured by ATKINS within the Barnwell Disposal Facility were reviewed. Based on the case of PE-HIC that was overpacked and disposed of in the Barnwell Disposal Facility, directions for how to dispose of PE-HIC at the domestic KORAD disposal site can be established in the future.

Keywords: Polyethylene, HIC, Barnwell disposal facility, Overpack

Analysis of Factors Affecting the Degassing Rate of C-14 in Biosphere Model

Youjin Oh, Ugyu Jeong, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

C-14 is a beta-emitting radionuclide with a long half-life and a key contributor to dose in radioactive waste disposal facility's radionuclide release scenarios. Therefore, proper biosphere modeling of C-14 is required for accurate safety assessment of disposal facilities. The pathways for C-14 include degassing from contaminated soil to the atmosphere, flowing through the vadose zone and aquifers via groundwater and pumped by well, or reaching surface water through bed sediments, followed by degassing. The C-14 degassed from soil and surface water to canopy layer is then used in the photosynthesis of plants and crops. Ultimately, human ingestion of these plants and crops is expected to be the main exposure pathway to C-14. Therefore, careful consideration of the degassing rate is essential for the biosphere modeling of C-14.

Two key factors influence the degassing rate in soil and surface water. First, an important factor affecting the degassing rate in the soil is the soil adsorption distribution coefficient (K_d) of C-14. According to the SKB R-13-46 report, the larger the K_d value, the lower the degassing rate is calculated, whereas RESRAD-ONSITE calculates a higher degassing rate with an increasing K_d value. These differences arise from the equations of the two models. SKB R-13-46 considers diffusion in the flux, while RESRAD-ONSITE assumes that the evasion flux rate is proportional to the level of soil contamination.

Secondly, the atmospheric isotopic exchange rate of C-14 is a significant factor influencing the degassing rate in surface water. When C-14-contaminated surface water exchanges carbon dioxide with the atmosphere, due to the very low concentration of C-14 in natural atmospheric carbon dioxide, C-12 dissolves into the surface water while C-14 is released into the atmosphere.

This study analyzed differences between models for soil degassing rates. Additionally, the isotopic exchange rate in surface water was examined by exposing solutions with different pH levels to the atmosphere. After achieving dynamic equilibrium of carbon dioxide, C-14 was added at 50 Bq/ml, and C-14 reduction over time was measured using a Liquid Scintillation Counter (LSC) and Total Organic Carbon Analyzer (TOC). Results showed rapid C-14 reduction in the solution with around pH 7, while in the solution with around pH 10, C-14 decreased slowly and even concentrated as the solution evaporated. The analysis and experimental data obtained from this study are expected to contribute to improving biosphere modeling of C-14 migration and safety assessment of disposal facilities in the future.

Keywords: C-14, Degassing rate, Isotope exchange rate, Biosphere modeling

Review on Dose Assessment Results for Clearance of Radioactive Metal Generated in Domestic Nuclear Power Plant

Yong Ho Jin*, Si Young Kim, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yongho.jin@khnp.co.kr

In Korea, the decommissioning of NPPs such as Kori Unit 1 and Wolsong Unit 1 is scheduled. Decommissioning generates a large amount of radioactive waste, and radioactive metal accounts for a significant amount of total radioactive waste. Therefore, if clearance of these radioactive metals is possible, it can be expected to reduce the volume of radioactive waste and reduce management costs. For clearance, it must be proven that the radiation dose and radioactive concentration criteria presented in the domestic law are satisfied. Therefore, this study investigated and analyzed the case of dose assessment for clearance of domestic metal waste generated during nuclear power plant decommissioning.

Seo et al. (KINGS) assumed the clearance of radioactive metals generated from the Kori Unit 1 reactor. Radiation source was considered Co-60, Cs-137, and Sr-90, which are the main nuclides in the metal waste of nuclear power plants, and its radioactive concentrations were set as clearance level (Bq/g). The RESRAD-RECYCLE computer code was used for the dose assessment, and the total of 28 scenarios were considered for the exposure scenarios. The result shows that if radioactive metal satisfies the clearance level, individual and collective dose criteria was also satisfied.

Choi et al. (KONES Corp.) assumed the clearance of the reactor vessel top head and insulation of Kori Unit 1. Radiation source was considered Fe-55, Ni-63, Co-60, and Ni-59, which considering the operating history of the nuclear power plant, and its radioactive concentration were set based on operating history including decay. The RESRAD-RECYCLE computer code was used for the dose assessment, and the total of 35 scenarios were considered for the exposure scenarios. The result shows that clearance of reactor vessel top head and insulation acceptable. The results of the dose assessment showed that the individual and collective dose criteria for clearance of the reactor vessel top head and insulation were satisfied. This study can be considered as a preliminary study for the dose assessment of clearance of radioactive metals.

Keywords: Clearance, Radioactive waste, Dose assessment, Metal

Review on Derivation Results of Clearance Level for Radioactive Metal Containing Uranium

Yong Ho Jin*, Si Young Kim, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yongho.jin@khnp.co.kr

If the radioactive concentration of each nuclide in radioactive waste is below the clearance level, radioactive waste can be managed as non-radioactive waste. The quantitative criteria for clearance are suggested in the law, but clearance levels are not provided for some nuclides. Therefore, cases are analyzed that deriving clearance levels for the clearance of radioactive metals containing uranium, a nuclide that does not have clearance standards.

Lee et al. (KNF) presented the clearance level of radioactive metals generated in nuclear fuel cycle facilities. Three nuclides, U-234, 235, and 238, were considered, and deterministic dose assessments were performed for melting and direct reuse recycling. Based on the most conservative dose assessment results, which are building reuse and scrap worker scenarios, clearance levels for surface contamination and radioactive concentrations were derived.

The Takeda et al. (JAEA) presented clearance levels for radioactive metals from nuclear facilities. Three nuclides, U-234, 235, and 238, were considered, and deterministic and probabilistic dose assessments were performed for melting and direct reuse recycling. Based on the most conservative dose assessment results, which is scrap worker scenario (inhalation pathway), clearance levels for radioactive concentrations were derived.

The European Commission presented clearance levels for radioactive metals generated during the decommissioning of nuclear facilities. Six nuclides, U-232, 233, 234, 235, 236, and 238, were considered, and deterministic dose assessments were performed for melting and direct reuse recycling. Based on the most conservative dose assessment results, which is the furnace scenario, clearance levels for surface contamination and radioactive concentrations were derived. The results of this study can be used for deriving clearance of radioactive metal containing radionuclides which the clearance levels are not suggested in the law.

Keywords: Clearance, Radioactive waste, Dose assessment, Uranium

Consideration of Source Definition on Radioactive Waste Analysis

Seung Uk Yoo*, Sung Hee Park, and Moonoh Kim

KONES Corp.

*yooswok@kones21.com

Recently, various studies have been actively performed on activated structures in nuclear power plant for preparing the decommission. The conventional method for analyzing activated structures involves using MCNP to calculate the neutron flux irradiating the structures, and then using ORIGEN to calculate the radionuclide inventory within the structures, as well as the gamma spectrum and intensity emitted by the activated structures. The calculated gamma flux and radionuclide inventory are then used to perform radiation shielding analysis for workers during the decommissioning.

However, if the source particle in the MCNP is changed from gamma (p) to decay gamma (SP), there is no longer required to use the gamma spectrum calculated by ORIGEN, which can simplify the conventional analysis process. Therefore, this study aimed to compare the results when using SP as the source particle with those from the conventional method to determine whether it could be effectively applied in the analysis of radioactive waste.

The subject of the analysis was bio-shield concrete that had been irradiated by a neutron flux of 10^{10} #/cm²·sec for approximately 40 years in a nuclear power plant, with the radionuclide inventory calculated one year after irradiation using ORIGEN. To compare the SP model, which uses SP as the source particle, with the ORIGEN model, which uses both the gamma-ray spectrum and radionuclide inventory from ORIGEN, it was assumed that gamma rays were emitted from activated concrete. The gamma-ray emissions were then calculated using MCNP, and the results were analyzed in terms of the energy spectrum.

The results showed that while the ORIGEN model calculated the gamma spectrum in accordance with the ORIGEN outputs, the SP model required the inclusion of key gamma-emitting radionuclides in the material information to produce results similar to those of the ORIGEN model. However, the SP model had the advantage of a significantly lower background, which allowed for the identification of peaks that might be obscured by background in the ORIGEN model.

To result out whether using SP as particle to analyze the radioactive waste can replace the conventional method or not, it requires much detailed analysis on radioactive waste. Due to the analysis that conducted in this paper calculated the results only with activated concrete, various radioactive waste requires to be analyzed for the future work.

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Keywords: MCNP, ORIGEN, Activation analysis, Radioactive waste

Improvement of Instrumentation and Control for the Evaporation Process

Yoon-Do Oh*, Hyun-Myoung Jung, Na-On Chang, Tack-Jin Kim, and Dong-Su Kim

Korea Atomic Energy Research Institute (KAERI)

*ydoh@kaeri.re.kr

The evaporation process handles and treats liquid radioactive waste generated from various facilities. This process is monitored through numerous instruments and operated through devices. The evaporation process is aimed at decontamination. The decontamination method involves three steps: first, evaporating low-level liquid radioactive waste using steam; second, condensing the evaporated gas into a liquid using cooling water; and third, further lowering the radioactive concentration through ion exchange. In this study, the reasons for improvement of the instrumentation and control (I&C) are explained. And post-improvement changes can be mentioned.

It is for the following reasons that I&C for the evaporation process is being improved. First, there are numerous indicators at the field in Radioactive Waste Treatment Facility (RWTF) whose values cannot be monitored in the control room. Regular field inspections have been conducted to observe the measurement points of the process, but these measurements are not continuously monitored and recorded. Instead, only the values observed by the operator during field checks have been recorded. Second, most of the equipment operates manually. The system is configured with relay-based control equipment, which can support automatic operation, but doing so requires many components. As a result, it is practically difficult to achieve automatic operation, increasing the risk of human error. Third, the aging of the process operating system. The RWTF has been operated since 1991, and the I&C for the evaporation concentration process is of the same age. This situation has increased the likelihood of malfunction in the I&C.

Due to the reasons mentioned above, we have initiated improvements to the I&C for the process. This improvement will enable the measurement and recording of instrumentation points that were previously unmonitored in the control room and will support automatic operation.

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Keywords: Evaporation, Radioactive waste treatment facility, Instrumentation and control, Liquid radioactive waste

Current Status of Radioactive Waste Filter Treatment

Ji-Hoon Lee^{1,*}, Kyung-Rok Park¹, Ji-Soo Yoon¹, Young-Hwan Hwang¹, Jung-Kwon Son¹, Moon-Oh Kim²,
and Na-Rae Lee²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²KONES Corp.

*leejihoon@khnp.co.kr

Untreated radioactive waste currently temporarily stored in Kori Unit 1 includes sludge, concentrated waste, and waste filters. Among them, waste filters have high radioactivity and are difficult to handle when they have been generated, so they have been stored in a temporary space in the nuclear power plant in an untreated state. In this study, Treatment technology applicable to radioactive waste filter of international application cases were investigated. In domestic nuclear power plants, the waste filter that occurred until the late 1990s was cement fixed and packaged in a 200 L steel drum, and as part of reducing radioactive waste, the high radioactive waste filters are stored for a long time to be treated after radioactive attenuation. DTS in the United States is applying APS (Advanced Polymer Solidification), a waste resin solidification technology, to solidify waste filters. APS is a representative waste stabilization method, and has been approved by the Conference of Radiation Control Program Directors (CRCPD) in the United States for the production of stable class B&C packages. The Nuclear Industry Radioactive Waste Executive (NIREX) in the United Kingdom suggests compression, solidification, grout, and crushing of waste filter treatment methods through Guidance Notes on the Packaging of Filters. France requires that waste filter wastes are be immobilized using a binding material. In addition, the packaging container used for the immobilization treatment is made of concrete with a thickness of 5 cm or more to ensure shielding performance. By being benchmarking the current treatment status of radioactive waste filters, it will be developed a technology that can secure the disposal suitability of radioactive waste filters temporarily stored in Korea in the future.

Keywords: Radioactive waste filter, Radioactive waste treatment

Review of Ion Exchange Resin Treatment Technology Development Trends

Seok-Ju Hwang*, Younghwan Hwang, Sung-Hoon Hong, and Jung-Kwon Son
Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)
*hsjhhhhh@khnp.co.kr

In the liquid radioactive waste treatment system of a nuclear power plant, ion exchange resins are used to move radioactive nuclides within the system to the solid ion exchange resin (IER) disposal. IERs must be solidified and sent to a disposal site according to the disposal site acceptance criteria.

In Europe, a study was conducted on the feasibility of applying IER Hot Isostatic Pressing (HIP), a technology for treating radwaste at high temperature and high pressure used in nuclear power plants. IER, a polymer material, was placed in a metal container stainless steel 304) for HIP and the thermal degradation of the polymer material was evaluated under high temperature and pressure operating conditions. In addition, experiments were conducted according to the drying condition of the IER and the conditions for adding additives to evaluate the solidification results according to the waste state and additives.

We plan to continuously conduct a review of the technical feasibility of applying IERs for treatment and disposal in domestic nuclear power plants in the future.

Keywords: Radioactive waste, IER (Ion Exchange Resin), HIP

Proposal of a New Idea for Screening Potential Errors in Radioactive Waste Data Based on Direct and/or Indirect Correlation of Radiological Parameters

Gun Young Yoon and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Radioactive waste must be comprehensively characterized in terms of its physical, mechanical, chemical, radiological, and biological characteristics from generation to disposal to ensure the safety of humans and the environment. Radiological characteristics are assessed through direct evaluation methods using radiation detectors and indirect evaluation methods such as scaling factors. Direct measurement methods may be prone to errors due to the measurement environment or human intervention. On the other hand, indirect measurement methods may face challenges in ensuring the reliability of calculated values under specific conditions.

To resolve these potential errors, validation and verification methods are essential to enhance the reliability of radioactive waste data. According to IAEA General Safety Requirements Part 5, regulatory body must ensure that the operator's activities meet safety and environmental protection requirements, and the characterization and classification of radioactive waste must be carried out according to the criteria established or approved by the relevant regulatory body.

In this regard, analyzing the correlations between reported radiological characteristics (dose rate, surface contamination, radionuclide composition, radioactivity concentration, etc.) provides a potential approach to predict the accuracy and reliability of such information. By analyzing correlations between these parameters, it is possible to detect potential errors in the reported data and identify patterns and discrepancies.

This study aims to propose verification methods by analyzing the correlations between various radiological parameters of radioactive waste. In this process, it analyzed the impact of factors such as the operational history of nuclear power plants on radiological characteristics, and it focuses on developing a screening approach to detect potential errors in radioactive waste data.

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Keywords: Radioactive waste, Radiological characterization, Direct correlation, Indirect correlation, Potential error, Screening

A Preliminary Study on Physical and Radiological Considerations of Soil and Building Radioactivity Measurement Systems for the Decommissioning of NPPs

Se Geun Lee* and Hyeok Jae Kim

Korea Research Institute of Decommissioning (KRID)

*sglee@krid.re.kr

For the safe and successful decommissioning after permanent shutdown of Kori Unit 1, it is one of the most important process that characterization of decommissioning wastes and on-site buildings. Radioactive soil can be generated during the decommissioning of nuclear power plants (NPPs) due to unexpected leakages from pipes, accidents such as spills, etc. It is of note that radioactive soil waste was generated and disposed from decommissioning of several foreign NPPs (e.g. Maine Yankee, Connecticut Yankee, José Cabrera, etc.). Those unexpected soil waste can cause much more decommissioning costs and manpower exceeding the original plan, and overload radioactive waste management and site release. In this aspect, proper radiological characterization based on radioactivity measurement should be conducted for determining the level of soil waste and the method of disposal accordingly.

Regarding site release, residual radioactivity concentration of on-site building is another key consideration. Site release, which is the final stage of decommissioning activity, should be accompanied by not only the methodology of demonstration of compliance, but also the dose criteria (i.e. 0.25 mSv/y) for release and reuse of building and site of nuclear power utilization facilities following the Notice of NSSC. On the other hand, radiological characterization such as radionuclide inventory including residual radioactivity concentration of building and site should be preceded before demonstrating the compliance of the dose criteria.

Considering perspectives of efficiency and safety of decommissioning activities, in-situ radioactivity measurement has been developed especially focused on radioactivity screening and rapid classification. On the other hand, to achieve safe and successful decommissioning with radioactive waste minimization and effective site release, it is necessary to apply more accurate measurement technology than conventional simple screening. Additionally, advances and developments of new technology also cannot be overlooked. Relatively latest technologies such as SLAM and 3D mapping based on the forth industry revolution (e.g. robotics, IoT, etc.) are being applied in this in-situ measurement field as well such as Createc's NV-Explore.

Moreover, referred to objectives of MARSSIM revision, the movement of foreign institutes for the standardization and regulation is expected to continuously apply improved and novel survey technology to the field of the NPP decommissioning. In terms of the decommissioning where proven technologies must be applied, following the NSSC Rule, this study aims to analyze the major considerations including both physical and radiological performances to apply accurate measurement purposes or robotics systems on radiological characterization for the decommissioning of NPPs.

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Keywords: Decommissioning, Site release, Radiological characterization, Radioactive waste management

Development of Soft Bag Design Requirements for Phase 3 Disposal Facility

Narae Lee* and Moonoh Kim

KONES Corp.

*nrlee@kones21.com

This paper presents the development of design requirements for soft bags intended for the disposal of very low-level radioactive waste (VLLW) in the third stage disposal facilities. The safe handling, transportation, and disposal of VLLW demand careful attention to safety and regulatory compliance due to the unique challenges posed by radioactive materials. Key considerations include material selection, structural integrity, size, capacity, handling, and adherence to regulations set by the International Atomic Energy Agency (IAEA).

Material selection is critical to ensure that the soft bags are durable and capable of safely containing VLLW. The bags should have a double-layer structure consisting of an outer skin made from materials that can withstand structural loads and environmental conditions, and an inner skin made from watertight materials to prevent leakage. All materials must meet safety standards to ensure containment during handling and transport.

The structural design of the soft bags must comply with IAEA regulations, including the ability to withstand a lifting force three times their maximum load without failure and to retain all contents if dropped from a height corresponding to the package's weight. These requirements ensure the bags remain intact and leak-proof in the event of accidents or mishandling.

Size and capacity should be optimized to accommodate the volume and characteristics of VLLW, allowing for efficient transport and storage. The design should be compatible with standard lifting and transport equipment while maintaining stability. Clear and durable labeling is necessary to identify the radioactive contents and ensure compliance with regulatory standards under various environmental conditions.

Safety measures are essential to minimize radiation exposure to workers and the public. The design must comply with all relevant national and international regulations, incorporating features to meet stringent transport and containment standards. Post-closure safety evaluations will also be considered to ensure long-term containment, minimizing potential risks to the environment and human health even after the disposal facility has closed.

It is believed that the soft bags developed according to these requirements will effectively perform multiple roles in radioactive waste management operations. They will ensure safe containment and transport of VLLW, prevent leakage of radioactive materials, withstand mechanical stresses and environmental conditions, facilitate handling and storage, and comply with all regulatory requirements. Additionally, they will provide long-term containment that meets post-closure safety evaluations, safeguarding human health and the environment throughout and after the disposal process.

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Keywords: Very low-level radioactive waste, Phase 3 disposal facility, Soft bag, Design requirements

A Preliminary Study on Technical Demonstration of In-Situ Gamma Spectroscopy Systems for Decommissioning Waste Characterization

Se Geun Lee*, Yong-Won Ma, and Do Yeob Kim

Korea Research Institute of Decommissioning (KRID)

*sglee@krid.re.kr

Radioactive waste management through radiological characterization of various types of decommissioning waste generated in large quantities over a short period of time is one of the most important key issue for successful and safe nuclear power plant (NPP) decommissioning. In particular, in the process of decommissioning domestic NPPs (e.g. Kori Unit 1 and Wolsung Unit 1), it is expected that the application of commercial in-situ technologies capable of rapidly measuring and classifying large amounts of decommissioning waste including soil to improve the safety of decommissioning workers and efficiently minimize decommissioning radioactive waste.

On the other hand, as opposed to the analysis of alpha- and beta-emitting radionuclides, which requires lab analysis due to the unique properties of the emitted radiation and the interactions of radiation and materials, the gamma-ray spectroscopy has the advantage of being able to be measured in-situ because of its relatively easy detection process. However, considering the high background of the decommissioning site and the generation characteristics of the decommissioning waste, which is mostly relatively low-level radioactive waste from clearance level to very low level, technical demonstration is essential to rely on the results of on-site gamma-ray spectroscopy.

In particular, unlike in-situ measurements such as ISOCS, which have been extensively studied for the purpose of site release, specific demonstration of on-site measurements for the purpose of radioactive waste measurements needs to be additionally conducted considering the decommissioning of Kori Unit 1. Therefore, this study investigated the technical demonstration methodology of the in-situ gamma-ray spectroscopy system for the preliminary classification of the level of structures, systems and components (SSCs) and the decommissioning wastes which subject to clearance at the decommissioning site. Moreover, this study aimed to present the essential considerations that system developers or users should take into account, such as the criteria for judging contamination, uncertainty and quality control, and quality assurance procedures.

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Keywords: Decommissioning, Digital twin, Radioactive waste management, Forth industrial revolution

Case Study on the Demonstration Elements of Waste Reduction Technology

Min-Gyu Choi*, DoYeong Mun, Yong Guk Son, and Moo-Keun Song

Korea Research Institute of Decommissioning (KRID)

*chomig@krid.re.kr

The Republic of Korea is preparing to decommissioning of Kori Unit 1 and Wolsong Unit 1, with an estimated generation of approximately 80,000 drums of radioactive waste during this process. Radioactive waste is produced not only during nuclear plant decommissioning but also from various operational activities within nuclear facilities. Consequently, various waste reduction technologies are being researched and developed to secure disposal site space and reduce disposal costs. Among these, induction heating and plasma torch melting technologies are efficient in treating waste due to their high volume reduction rate.

However, these technologies require thorough demonstration for radioactive waste treatment. For example, before applying plasma torch melting equipment at the decommissioning site, the Kozloduy Nuclear Power Plant in Bulgaria assessed equipment components, normal and abnormal operational conditions, and conducted simulated waste melting tests. Similarly, in the Republic of Korea, induction heating systems for radioactive waste have undergone evaluations of system integrity and processing tests using simulated waste.

This paper investigates demonstration cases of these electrical and thermal waste reduction technologies and considers the elements of demonstration tests necessary for applying domestic treatment technologies to nuclear plant decommissioning.

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Keywords: Radioactive waste, Waste treatment, Volume reduction, Demonstration, Melting

Integrated Approaches to Analyze Irradiated Materials in Hot Cell Facility of KRID

Seoha Kim, Junghyun Park, Seung-Ho Lee, and Yeong Su Ha*

Korea Research Institute of Decommissioning (KRID)

*zerowater@krid.re.kr

For the progress on nuclear waste management, it is imperative to perform integrity assessments and radiochemical analyses of irradiated substances. It requires facilities with high shielding functions which can accommodate intermediate radioactive materials and various forms of specimens. Despite the necessity, the laboratories are absent, and therefore the accessible database are restricted. Therefore, Korea Research Institute of Decommissioning (KRID) aims to develop the technologies and to set new benchmarks in the field of nuclear decommissioning and waste management.

Hot cell of KRID is currently constructed and designed to enhance the management of radioactive materials. The facility will consists of comprehensive cells specialized in material characterization and analytical capabilities required for investigation of irradiated materials. It will be composed of seven cells which are transport cart cell, sample processing cell, sample preparation cell, characterization analysis cell, pretreatment cell, acceptance/storage cell and maintenance cell, and intended to minimize radiation exposure for personnel. Additionally, the experiments can be conducted remotely in the cells.

In this study, we concentrated on the operational strategies and specifications of internal equipments. Furthermore, the research encompasses detailed scenarios for the unloading from truck bay, moving routes in cells and experimental methods using manipulators, overhead cranes, gripper and padirac. In addition, we optimized detailed movement paths of equipments through hatch doors and materials through pass-thru doors and pneumatic tube system. As a result, we considered the problems and interferences in operational methods and devised solutions.

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Keywords: Hot cell, Nuclear waste management, Manipulator, Padirac

A Case Study on Radiochemical Analysis of Non-Fuel Materials in the SFP of PWR

Kyungho Roh*, Jinwon Son, and Kiyoun Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*rohkyungho@khnp.co.kr

To dispose of intermediate and low-level radioactive waste, more than 95% of the radionuclides present in the packaging must be identified according to the Nuclear Safety Commission Notification No. 2021-26. This includes identifying H-3, C-14, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Tc-99, I-129, Cs-137, Ce-144, and gross alpha. Most of these radionuclides belong to difficult-to-measure radionuclides that emit beta radiation or low energy gamma rays and X-rays. To quantify their activity concentrations, chemical separation of the radionuclides may be required to eliminate or minimize the effects of interfering substances or nuisance radionuclides. The choice of preprocessing, target nuclide and interference nuclide chemistry separation, sample preparation for measurement, and analytical methods is made based on the radiochemical properties and chemical characteristics of the target nuclide. Quantitative analysis of non-gamma emitting alpha or beta emitters, also known as difficult-to-measure radionuclides, requires chemical separation of the target nuclide using various chemical techniques such as precipitation, ion exchange, solvent extraction, and chromatography. Non-fuel wastes can be classified into metallic and non-metallic materials. Common material components of metals include stainless steel alloys, silver-indium-cadmium alloy, Inconel, zircaloy, and zirlo, while common material components of non-metals include boron carbide, alumina, and pyrex. These wastes contain different radionuclides depending on the composition of the material component due to irradiation. This paper reviews existing representative analytical methods for three major nonvolatile difficult-to-measure radionuclides (tritium, carbon-14, and iodine-129) to derive necessary information for analyzing radionuclides contained in metal components of non-fuel materials within spent fuel pool. Based on the derived results, validation of the findings will be conducted by performing actual analyses of irradiated non-fuel materials at an analysis facility.

Keywords: Non-fuel material, Difficult-to-measure radionuclides, Nonvolatile

Development of Radioactive Waste Clearance Procedures for Large Square Concrete Drums Generated at Kori Nuclear Power Plant

Woo Nyun Choi^{1,*}, Kwangsoo Park¹, ChungSik Lim¹, Changkyu Kim¹, HaeWoong Kim¹, Donghun Park¹,
Taeseob Lim¹, Young Hwan Hwang², and Sunghoon Hong²

¹Doosan Enerbility

²Korea Hydro & Nuclear Power (KHNP)

*woonyun.choi@doosan.com

Concentrated and resin cement-based solidified wastes generated during the operation of the Kori Nuclear Power Plant (NPP) from 1987 to 1997 are packaged in 200 L drums and stored for long periods in large square and circular concrete drums. The total number of square concrete drums is 820, and the total weight of the concrete wastes except for 200 L drums in the concrete drum is conservatively estimated at about 3,368 ton. In order to determine whether self-disposal is possible, the waste must be below the allowable concentration for self-disposal and satisfy the allowable dose for self-disposal (individual dose less than 10 $\mu\text{Sv/y}$ and collective dose less than 1 $\text{man}\cdot\text{Sv/y}$) according to Nuclear Safety and Security Commission notice No. 2023-7.

The contamination status of the square concrete drum may vary depending on whether the inner 200 L drums are damaged. Therefore, the procedure for evaluating the separation of contamination and non-contamination in the concrete drum and self-disposal of the non-contamination area was developed regardless of whether the 200 L drums are damaged. First, to determine whether there are any hot spots, the top (1 piece) and sides (4 pieces) of a square concrete drum are cut into plate-shaped concrete. After moving the plate-shaped concrete to the contamination measurement location, the surface dose rate (within the natural radiation level) and contamination (Alpha 0.04 Bq/cm^2 , Beta 4 Bq/cm^2 or less) are checked to determine whether a hot spot exists. If a hot spot is identified, it is decontaminated using scabbling equipment, and for plate-shaped concrete judged to have no hot spot, on-site gamma nuclide analysis is performed using ISOCS. For cross-validation of ISOCS results and beta/alpha nuclide analysis, three or more samples are collected from the sample group and nuclide analysis is performed. Based on the results of the nuclide analysis, the satisfaction of self-disposal allowable doses is confirmed through the calculation of the RESRAD code, and finally, the possibility of self-disposal is derived.

The concrete waste self-disposal assessment procedure developed through this study can be used in the future disposal process of 820 large square concrete drums. By minimizing the amount of non-radioactive concrete waste classified as radioactive waste, disposal costs can be reduced and free space in disposal site can be provided.

Keywords: Concrete waste, Self-disposal, Kori NPP, ISOCS, RESRAD

Cumulative Dose Assessment of Decontaminated Soil From Uranium Conversion Facility Decommissioning Site in KAERI

YuJeong Choi*, Sang-mo Hwang, Sung-Jin Han, Kyung-min Kim, and Won-Hyuk Jang

Korea Atomic Energy Research Institute (KAERI)

*choiyj@kaeri.re.kr

The Korea Atomic Energy Research Institute is continuously reducing radioactive waste through self-disposal, and self-disposal is expected to become more active due to the advantages of cost reduction and reduction of generated radioactive waste. Since clearance level radioactive waste generated in this way is sent to a limited waste disposal site, the cumulative dose according to the continuous treatment of clearance level radioactive waste must be confirmed.

The Korea Atomic Energy Research Institute is researching a decontamination method for uranium-contaminated soil generated from the Uranium Conversion Facility. Accordingly, clearance level radioactive waste of decontaminated uranium-contaminated soil is expected to continuously be generated.

In this paper, if the continuously generated uranium-contaminated soil keep buried in a specific landfill, the cumulative dose for the next five years is evaluated, and the total amount of waste that can be buried according to the currently in-progress decontamination method is calculated, and it is confirmed that the cumulative dose of the waste does not exceed the dose limit.

The dose assessment was performed using RESRAD-Onsite 7.2, and the dose assessment was performed by reflecting the regional characteristics and population near the landfill.

As a result, the maximum annual individual radiation exposure dose rate was 9.4 $\mu\text{Sv/y}$ (with 800 ton/yr, not feasible amount, max 100 kg/yr), which was lower than the self-disposal allowable concentration. In addition, if 100 kg of waste was landfilled every year based on the current decontamination experiment concentration, it was confirmed that the cumulative dose after 5 years would satisfy 10 $\mu\text{Sv/y}$, which is the self-disposal allowable dose according to the Nuclear Safety and Security Commission notice. The Maximum dose rate occurred after 600 years later, due to the behavioral characteristics of uranium.

Advancing Nuclear Waste Management: Development of KRID's Comprehensive Hot Cell Facility

Junghyun Park*, Seoung-Ho Lee, Seoha Kim, Seokhan Kim, and Yeong Su Ha

Korea Research Institute of Decommissioning (KRID)

*JH@krid.re.kr

The Korea Research Institute of Decommissioning (KRID) is advancing its nuclear waste management efforts by developing a state-of-the-art Hot Cell facility to support the property analysis of Kori Unit 1 and Wolsong Unit 1 nuclear power plants. This initiative includes the establishment of a comprehensive Hot Cell system designed to analyze the physical and radiochemical properties of intermediate and low-level radioactive waste.

Nuclear decommissioning has been conducted in only four countries worldwide. Consequently, the available database are very limited. To address this issue, KRID aims to advance the physical and radiochemical analysis of radioactive waste through the development of its Hot Cell facility.

The Hot Cell facility currently under construction at KRID will consist of seven cells, each designed to perform a range of critical functions including sample processing, pre-treatment, physical property analysis, and chemical analysis pre-treatment. This multifunctional system will enable the handling of complex waste materials, conduct detailed analyses, and provide essential data necessary for effective waste analysis and management.

KRID's Hot Cell system is expected to significantly enhance the understanding of waste characteristics, improve management and processing procedures, and contribute to compliance with safety and environmental regulations. By developing this advanced analytical capability, KRID aims to serve as a key hub in Korea's nuclear waste management network and foster collaboration and knowledge sharing within the industry.

This project not only represents a substantial advancement in Korea's nuclear waste management capabilities but also addresses challenges related to waste characterization and management, contributing to the global knowledge base. KRID is committed to advancing waste management practices and promoting sustainable environmental management through these efforts.

Acknowledgements

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Keywords: Intermediate-level radioactive material, Waste management, Hot cell, KRID

Development of a Point Sampler for Efficient In-Tank Sampling Inaccessible to Workers

Sangbok Lee*, Minho Kwak, and Seungil Kim

Neosiskorea Co., Ltd.

*sblee@neosiskorea.com

Kori unit 1 and Wolsong unit 1 have reached shut down the end of service life and have been permanently. As more nuclear power plants will be expected to decommission in the future, it is important to radiological characterization by establish effective work plans and calculate worker exposure doses. When dismantling the targets of the work are various facilities of various shapes, including tanks storing radioactive waste. Small tanks will be expected to be easier to sampling, but medium and large tanks will be expected to be difficult to sampling. To overcome these difficulties, this study aims to develop a point sampler that can effectively sampling from inaccessible tanks to facilitate radiological characterization.

The point sampler was manufactured in an assembled form for more ease of human handling. The guides were manufactured in 1 meter increments per guide to make assemble them and distinguished the depth, a ball valve and wire are used by the front of the sampler to collect can be sampling at the desired depth. In addition, a supporter was made to worker in convenience and safety.

To test the effectiveness of the point sampler, sampling tests were performed on a 15 m height and 12 m width storage tank. Samples were collected from 10 systematic random point in the storage tank, each sampling had taken approximately 30 minutes with four operators and during the work minimizing contact with the sample and without cross contamination. During the sampling the sample volume taken consistent at 850~900 mL, it was able to sample both liquid and sludge.

The importance of optimal work planning through radiological characterization for decommissioning work as nuclear power plants is becoming increasingly important in these days where worker safety is critical. The developed point sampler will be provided an effective way to sampling tanks that are difficult to access, more reliable radiological characterization and optimal work planning.

Keywords: Systematic random sampling, Radiological characterization, Radiation work planning, Point Sampler

Analysis of the Immobilization Mechanisms of Technetium and Iodine in Cementitious Materials via Co-Precipitation Experiments

Yejin Kim and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Cement is the most commonly used material for the solidification of radioactive waste, and understanding the nuclide behavior within cement phases is important for identifying the main mechanisms of radionuclide immobilization. During cement hydration, various minerals can be formed which contribute to the retention properties of nuclides. Calcium silicate hydrate (C-S-H) gel, one of the major cement hydration products, comprises a large portion of the cement matrix and is effective in immobilizing radionuclides through electrostatic adsorption and ion exchange with calcium ions. Ettringite ($\text{Ca}_6\text{Al}_2(\text{SO}_4)_3(\text{OH})_{12}\cdot 26\text{H}_2\text{O}$), which forms early in cement hydration (~7 days), is also effective in capturing anionic radionuclides through ion exchange with sulfate (SO_4^{2-}).

Adsorption experiments on these minerals are actively being conducted, however, these studies typically focus on surface reactions occurring after the complete formation of the minerals, without considering the reactions that occur during the actual hydration process. Furthermore, since the relative amounts of minerals can change due to decomposition and synthesis during hydration, it is important to clarify which minerals are more effective for nuclide immobilization for a long-term disposal perspective. Therefore, incorporation mechanisms during the formation process of the cement solidified specimen should be investigated.

The interaction mechanisms between waste forms and radionuclides vary with ionic charge. For instance, C-S-H adsorption experiments show that cations are absorbed significantly more than anions, due to the negative surface charge of the C-S-H influenced by SiO_2 . In case of anions, sorption on C-S-H can be negligible, however, previous experimental results show high leachability indices (8-9) in cement solidified specimen. This suggests that other minerals, such as ettringite, may affect nuclide immobilization. Therefore, in this study, the immobilization mechanisms of C-S-H and ettringite according to the type of nuclides were analyzed by co-precipitation experiments.

Co-precipitation experiments were conducted in triplicate on ettringite and C-S-H with iodine and rhenium (a surrogate for technetium). Cesium and strontium were also performed to compare the uptake trends with anions. The minerals were synthesized via aqueous precipitation, and stock solutions of non-radioactive nuclides at various concentrations (0.1-10 mmol/L) were used to initiate solid precipitation. The concentrations of the solutions after the reaction were analyzed by ICP-OES, and phase analysis, including XRD, was conducted to examine the formation of minerals. As the result of this study, the immobilization preferences of various nuclides for each mineral can be verified, providing fundamental data for understanding leaching behavior in cementitious waste forms.

Keywords: Calcium silicate hydrate, Ettringite, Co-precipitation, Cement solidified waste, Radionuclide immobilization

Methodology for Efficient Disposal of the Fluorescent Liquid Waste Generated From the Operation of LSCs in NPPs

Hang-Rae Cho^{1,*}, Jun-Seok Nam², Seok-Ju Hwang¹, Ji-Soo Yoon¹, and Jung-Kwon Son¹

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Korea Hydro & Nuclear Power (KHNP)

*hrcho6795@khnp.co.kr

The Liquid Scintillation Counters (LSCs) are widely used to measure beta-emitting nuclides, such as tritium (^3H) and radioactive carbon (^{14}C), in the liquid samples of nuclear power plants (NPPs). Due to the measurement principle of LSC, which detects and counts visible light emitted from a flash phenomenon by mixing a fluorescent cocktail consisting of solvents and scintillators into a liquid sample, vials of sample mixed with fluorescent liquid are generated as fluorescent liquid waste after measurement.

The fluorescent liquid waste is collected and stored for a long time within the plant radiation control area (RCA) and then disposed through a separate technical service project with a lot of cost, time and licensing processes because it cannot meet the acceptance criteria of Gyeongju Radioactive Waste Disposal Facility.

We reviewed the current methodologies to dispose of the large amount (170 drums) of fluorescent liquid waste collected over a long period of time and studied the technology and application methods for immediately disposing of the small amount of fluorescent liquid sample vials generated daily using modular small device in a near compartment equipped with an exhaust line as soon as possible without collecting and storing in bulk them.

Keywords: LSC, Liquid scintillation, Immediate disposal, Fluorescent liquid waste, Fluorescent cocktail

Review on the Decontamination Performance of Radioactive Liquid Waste Processing System in Domestic NPPs

Hang-Rae Cho^{1,*}, Ji-Soo Yoon¹, Seok-Ju Hwang¹, Hyo-Jin Park², and Jung-Kwon Son¹

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Korea Hydro & Nuclear Power (KHNP)

*hrcho6795@khnp.co.kr

The number of operating nuclear power plants (NPPs) has been gradually increasing with the new construction and continuous operation policy for the future electric energy demand in Korea. The radioactive release into the marine environment from each NPP needs to be managed below the detection limit as much as possible, considering the number of units and public acceptance, through more thorough processing of radioactive liquid waste.

The radioactive liquid waste processing system of domestic NPPs has evolved in accordance with the flow of technological development, starting with the evaporator, passing through ion exchange demineralizer such as selective ion exchange system (SIES) and liquid radwaste demineralizer package system (LRDPS) with the pretreatment subsystems such as centrifuge, submerged hollow fiber microfiltration (MF), spiral wound nanofiltration(NF) and reverse osmosis (RO) process, and up to the RO package system including MF, RO process, and ion exchange polisher.

As the main decontamination process, the evaporator has the best performance and no additional performance improvement is required. In the case of LRDPS and RO package, nuclides such as Co-58 and Sb-124,125 tend to remain in trace amounts at the outlet of the final ion exchange process due to the colloidal behavior or low ion selectivity.

We reviewed the characteristics of membrane separation and ion exchange technologies applied to the radioactive liquid waste processing system and investigated the ways to highly improve the decontamination performance of the system by filtering selectively and physically the problematic nuclides in the final process.

Keywords: Decontamination performance, Ion exchange process, Filtering selectively and physically

Introduction on Overseas Plans to Dispose of High Level Vitrified Form at Deep Geological Repository (DGR) After Reprocessing SF and Vitrifying HLW

Cheon-Woo Kim*, Sunghoon Hong, and Young Hwan Hwang

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*cheonwoo@khnp.co.kr

Nuclear fuel reprocessing refers to a technology that dissolves spent fuel in nitric acid, etc., and then extracts useful uranium and plutonium. Since fuel can be obtained again through reprocessing, France, United Kingdom, Japan, India, China, Russia, etc., are planning to reprocess or operate reprocessing facilities. Except for uranium and plutonium, which will be used for nuclear fuel, the remainders are high level radioactive waste (HLW). It is melted at about 1,000 degrees celsius using vitrification technology, stabilized, and poured in a metal canister for disposal. In France, after reprocessing SF, liquid radioactive waste is being vitrified at the La Hague reprocessing facility and then disposed of at the Cigeo in Bure, a 500 m underground disposal site. Previously, in the case of Japan, SF was reprocessed in France and UK to return the vitrified canister, and in the future, it is planned to be vitrified at the Rokkasho reprocessing facility and then disposed of DGR. In China, SF is also planned to be reprocessed and disposed of, and Beishan is the most likely DGR. In this study, we would like to introduce examples of several foreign countries that vitrify liquid radioactive waste after reprocessing SF and then finally dispose it at the DGR.

Keywords: High level radioactive waste, Reprocessing, Spent fuel, Vitrification technology, Vitrified form, Deep geological repository

Development of Tritium and Helium-3 Recovery Technology From Gaseous Waste of Gaseous Tritium Light Sources (GTLS)

Jiseon Kim, Woo-in Lee, and Soon-hwan Son*

AGENCORE CO., LTD.

*shson@agencore.net

The purpose of this study is to develop a technology for effectively recovering and separating tritium and helium-3 (^3He) from inoperable and out of life gaseous tritium light sources (GTLS). The tritium and helium-3 recovery system consists of a crushing subsystem and a tritium and helium-3 separation and recovery subsystem. The crushing part includes a glass crusher for breaking borosilicate 3.3 glass and a buffer tank for collecting the mixed gases released from the crushed glass. The key components of the tritium and helium-3 separation and recovery part include a molecular sieve (MS) bed, cryo MS bed, and metal getter bed. The MS bed operates at room temperature to remove particulate impurities from the gas mixture collected in the buffer tank after crushing. The MS bed is filled with Molecular sieve type 3 A (0.3 nm). The cryo MS bed operates at -196°C to remove gaseous impurities, excluding tritium and helium-3. This cryo bed uses a cryopump of Molecular sieve type 4 A of 300 g. The metal getter bed employs a ZeFe (Zirconium-Iron) getter, which selectively absorbs tritium from the tritium/helium-3 gas mixture, with a storage capacity of 77.7 GBq/g. The ZeFe metal getter content of the bed is 500 g. The tritium/helium-3 recovery process developed in this study is planned to be demonstrated its effectiveness and safety through verification experiments using tritium in the future.

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Keywords: Tritium, Gaseous Tritium Light Sources (GTLS), Helium-3, GTLS recovery

Validation of EDTA Measurement in Radioactive Waste Using the Leaching Method

Hye Jin Choi, Gye Cheon Choi, Heung Rae Lee, and Changryong Lee

Korea Standards Anal. Corp.

*hyejin.choi@ko-sa.co.kr

This study aims to validate the effectiveness of a method for analyzing the content of chelate agent, specifically EDTA (ethylenediaminetetraacetic acid), remaining in the dry active waste (DAW). To ensure the reliability of EDTA analysis, validation methods for five key parameters were determined, and evaluation criteria were established.

During the analysis of EDTA, peaks were observed before and after pretreatment, confirming their presence at a wavelength of 545 nm (Zr-EDTA peak). The linear correlation (R^2 value) of the measured values with the added EDTA standard, which serves as a recovery benchmark, was found to be 0.997.

To verify the accuracy of the EDTA separation and quantification method by leaching, the recovery rate of the standard substance added before leaching was measured, resulting in an average recovery of 89.07% with a relative standard deviation (RSD) of 0.4%

Cermet Waste Forms for Immobilization of High Heat-generating Nuclides

Byeonggwon Lee, Ga-Yeong Kim, Jung-Hoon Choi*, Ki Rak Lee, Hyun Woo Kang, and Hwan-Seo Park

Korea Atomic Energy Research Institute (KAERI)

*mrchoijh@kaeri.re.kr

Cermet (ceramic-metal) materials are heterogeneous composites composed of a ceramic phase and a metal phase. Although various studies have been conducted on cermet materials, research on their application for the immobilization of radioactive waste has been relatively insufficient. Cermet-based waste forms can immobilize radionuclides within a chemically stable ceramic phase that exhibits excellent leach resistance and encapsulate the ceramic phase with a metal matrix. The metal acts as a protective barrier, thereby minimizing the exposure of radionuclides in disposal environments. In particular, these cermet waste forms possess high thermal conductivity due to the continuous metal phase, enabling the effective dissipation of heat generated during the decay of high heat-generating radionuclides. This allows the waste form to maintain its stability over the long term and prevents various issues caused by decay heat. In this study, we immobilized the representative high heat-generating radionuclides, Sr and Cs, by incorporating them into the SrTiO_3 and Cs pollucite phases, respectively, and subsequently fabricated these materials into a cermet waste form. The physicochemical properties and leaching characteristics of the fabricated waste form were evaluated, confirming the feasibility of cermet materials as waste forms for the immobilization of high heat-generating radionuclides. This suggests that cermet materials could contribute to the reduction and safe disposal of radioactive waste.

Keywords: Cermet waste form, Ceramic waste form, SrTiO_3 , Cs pollucite

Removal Technologies for Volatile Radionuclides ^{14}C and ^3H in Organic Compounds of Radioactive Waste

Dong-Cheol Kim*, Hui-Gyeong Kim, Keun-Teag Park, Min-Young Jung, Mun-Su Han, and Eun-Kyung Jang

DAON Technology Co., Ltd.

*pvs98ko@naver.com

In domestic nuclear power plants, the typical organic compounds generated include spent activated carbon, spent adsorbents, contaminated decontamination materials (which are classified as solid waste), waste lubricating oil, and spent fluorescent liquids (classified as liquid waste). Among these organic compounds, waste lubricating oil is disposed of through a refining (self-disposal) process. However, the disposal of the other organic compounds is not possible due to the lack of commercial disposal technologies, and they are currently being stored within the power plants.

The organic compounds contain gamma-emitting radionuclides such as ^{60}Co , ^{137}Cs , and ^{95}Nb , as well as volatile beta-emitting radionuclides like ^{14}C and ^3H . To reduce the volume of radioactive waste and disposal costs, these radionuclides should be separated from the waste through physical and chemical treatments. The separated radionuclides need to be stabilized to meet the requirements for disposal suitability.

This study aims to review the removal technologies for volatile beta-emitting radionuclides ^{14}C and ^3H present in solid and liquid organic compounds.

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Keywords: Organic compound, Self-disposal, ^{14}C , ^3H

A Study on Disposal Methods Using Heavy Water Reactor Waste Adsorbent Characteristics

Minyoung Jung, Dongcheol Kim, Munsu Han, Huigyeong Kim*, and Keunteag Park

DAON Technology Co., Ltd.

*E.kimhphp@naver.com

Unlike light water reactors, heavy water reactors have steam recovery systems such as reducers, cooling systems, and heat transfer systems, and the steam recovery system in heavy water reactors uses adsorbents (zeolites) to remove water. A range of radionuclides, such as C-14 and H-3, are adsorbed on waste adsorbents, and the adsorbed nuclides (C-14 and H-3) are characterised by easy volatilisation at temperatures above about 100°C and are easily soluble in water, therefore, stabilization technology that enhances the physical and chemical stability of radionuclides adsorbed on waste adsorbents is essential to ensure their disposal suitability.

Zeolites are used in a wide range of industries, especially in environmental applications and as adsorbents to remove impurities such as CO, CO₂ and H₂O in gas purification processes to produce ultra-pure gases. Zeolites have a porous structure that can accommodate a range of cations. It is composed mainly of aluminosilicates, which are made up of silicon, aluminium and oxygen, making it extremely durable and resistant to radiation. In addition, it has the advantage of applying the purification process at a relatively low temperature of about 300°C in regeneration temperature compared to activated carbon.

However, considering the satisfaction and cost of domestic disposal suitability, waste adsorbent waste used as a desiccant for heavy water reactor nuclear power plants has a large volume to dispose of, and due to the low-temperature volatilization of adsorbed radionuclides, it plans to dispose of them when the treatment technology is secured, and it is currently temporarily stored in nuclear power plants, so a solution is needed.

Therefore, this study aims to evaluate the effect of various variables on the adsorption rate based on the characteristics of zeolites and radionuclides, and to propose a method for radionuclide removal from zeolites that meets the requirements of volume reduction and disposal suitability.

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Keywords: Heavy water reactor, Waste adsorbent, Radionuclide, Disposal suitability

A Simplified Comparison of Safety Assessments for LLW Vault Disposal and VLLW Landfill Trench Disposal Using AMBER Code

Su Yeon Lee and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Radioactive waste is generated throughout the lifecycle of a nuclear facility, including during operation and decommissioning. Radioactive waste is finally disposed of after an appropriate treatment process for each level according to the radioactive concentration, type of nuclide, and heat generation rate. Consequently, as the demand for nuclear power plants increases or decommissioning takes place, the amount of radioactive waste generated and the demand for disposal facilities also increases. The operator of a radioactive waste disposal facility is responsible for demonstrating its safety. IAEA Specific Safety Requirement No. 5 states that the operator shall carry out a safety assessment and develop and maintain a safety case. According to the IAEA Glossary, a safety assessment is defined as an assessment of all aspects related to protection and safety. The operator shall carry out the safety assessment at all stages of the disposal facility, and the results of the safety assessment shall be thoroughly documented.

Radioactive waste disposal facilities are designed in various forms according to the characteristics and types of waste to be disposed of, and they also provide shielding and containment of radioactive materials using engineered barriers such as vaults or trenches. It is important to choose the appropriate disposal strategy according to the type of radioactive waste. Radioactive waste classified as Low Level Waste or lower is disposed of in near-surface disposal facilities. The IAEA proposed a safety assessment methodology for Near Surface Disposal Facilities in the Improvement of Safety Assessment Methods for Near Surface Disposal Facilities (ISAM) document by conducting a joint research project. In this document, the previously proposed safety assessment methodology was reviewed and improved, and vault test cases were provided. IAEA-TECDOC-846 compares of country-specific methodologies developed for long-term safety assessments of near surface disposal facilities and provides a trench test case.

In this study, AMBER software is used to compare the ISAM vault test case and the IAEA-TECDOC-846 trench test case. The two test case conditions are applied to the same radioactive waste, the dose and transfer flux for each period are calculated and compared with each other, and the differences are analyzed. The results of this study can be used as basic data for research on near-surface disposal by analyzing the characteristics and differences of radioactive waste disposal facilities.

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Keywords: Disposal, Radioactive waste, Safety assessment, Trench test case, Vault test case

Comparison and Analysis of Clearance Systems for Low Activity Waste in Selected Countries

Hee Won Lee and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

According to the IAEA PRIS, the number of permanently shut down nuclear power plants is continuously increasing, and the waste generated from this process is predominantly low active radioactive waste. Low active waste includes clearance waste, which, in radiological terms, refers to a level at which regulatory control is not required a term that is internationally recognized. Clearance is an effective disposal option for reducing radioactive waste, which is why international organizations have issued related safety standards and guidelines, and many countries have incorporated these into their legal frameworks. While most countries adopt similar clearance systems by reflecting the safety standards and guidelines of international organizations, some countries have developed unique clearance systems tailored to their national characteristics.

In the United States, due to the relatively low disposal costs, very low-level radioactive waste is disposed of according to US NRC 10 CFR 20.2002. Although clearance has not been legislated into the national laws, the ANSI/HPS N13.12-2013 document presents surface and volume radioactivity standards for clearance. While this document has not been incorporated into official national policy, it is significant in that it provides clearance levels for both volume and surface.

In Germany, the clearance system is applied independently according to the country's specific characteristics. Unconditional clearance is applied not only to solid materials but also to liquids, with clearance levels provided for both volume and surface. Additionally, specific clearance is applied to particular situations such as landfill or incineration of solid materials, buildings for demolition, and metals for recycling, with corresponding clearance levels specified. Germany operates clearance measurement facilities, and generally, 100% of the clearance materials are measured.

Japan applies the clearance system independently according to its specific characteristics. In 2020, with the establishment of regulations on clearance, clearance levels were defined for 257 radionuclides specified in GSR Part 3, as well as for 17 radionuclides independently defined by the NRA. Additionally, clearance evaluation unit was increased from 1 ton to 10 tons. Japan operates clearance measurement equipment to assess whether clearance is possible, and waste is disposed of based on the approval of the NRA.

This study analyzed countries that have uniquely implemented clearance systems tailored to their national characteristics. In the future, a more detailed analysis of the gaps between different countries will be conducted. This study could contribute to establishing an optimal clearance system for the efficient disposal of radioactive waste and provide direction for future research.

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Keywords: Clearance, Clearance level, Unique, Germany, Japan

Specific Radioactivity Analysis of Activated Metal Using Gamma Spectrometry System and Gamma Counter

Sangrok Kim* and Suhyoung Lee

Korea Institute of Radiological and Medical Sciences (KIRAMS)

*kim@kirams.re.kr

The Korea Institute of Radiological and Medical Sciences has been producing radioisotopes for over 30 years using 30 MeV and 50 MeV cyclotrons, which has resulted in the generation of various types of activated metal waste. Radioactive waste disposal methods are limited to either self-disposal or consignment disposal to the government, with specific radioactivity being the key criterion for determining the appropriate method. In this study, target back supports were selected from the various activated metals, and their specific radioactivity was measured.

The supports were chosen as the subjects of analysis. Since these metals were used in an unsealed radioisotope production facility, they were washed before measurement to prevent contamination of the measuring equipment. The supports were then grouped based on surface radiation measurements using a contamination meter. As the supports were in plate form, a hydraulic punching machine was used to sample them into coin-shaped pieces with a diameter of 6.5 mm. To assess potential variations in specific radioactivity based on sampling location, samples were collected from various points on each support. The samples were analyzed using a gamma spectrometry system calibrated with certified radioactive material produced by the Korea Research Institute of Standards and Science. Additionally, the same samples were analyzed using a gamma counter to evaluate the feasibility of specific radioactivity analysis with this method.

The 28 supports were divided into two groups according to surface contamination levels, and three samples per support were measured for 30 minutes and 10 minutes, respectively, using both the gamma spectrometry system and the gamma counter. The weight of the individual samples ranged from 0.84% to 1.56% of the total sample weight, and all samples exhibited specific radioactivity levels exceeding the self-disposal standards by factors of 18 to 1,647. At the center of the supports, ^{60}Co , ^{207}Bi , and ^{231}Th were detected, while at locations further from the center, the radioactivity of ^{207}Bi diminished or was undetectable. The gamma counter detected only ^{60}Co , with a measurement efficiency consistently ranging from 13% to 15% of that of the gamma spectrometry system.

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Keywords: Radioactive waste, Self-disposal, Specific radioactivity, Activation, Gamma-spectrometry

Verification of Marine Accident Data From mTIS and Derivation of Marine Accident Frequency for Radiological Risk Assessment in Maritime Transport

Yoo-Kyoung Shim¹, Yoonsun Chung¹, Kiwon Jang², Ara Go², and Han-Ki Jang^{3,*}

¹Hanyang University

²Korea Institute of Nuclear Safety (KINS)

³Korea Association for Radiation Application (KARA)

*hkjang@ri.or.kr

Various studies have been conducted to evaluate the radiological risk assessment for transporting radioactive waste by sea. IAEA summarized international research including the U.S., Japan, and France and analyzed their methodology in TECDOC-1231. According to this document, most foreign studies employed data from U.S. Lloyd's which could not be fully suitable for Korea with event tree. However, adopting this method has restrictions since the U.S. data could not be fully suitable for Korean environment, and applying the event tree method is limited as the Korean marine accident category has a dependency on the importance of severity.

Instead, Korean-specific marine accident data and the grid method are considered to be more appropriate for risk assessment in order to avoid dependency on accident categories. To utilize the statistical analysis system of the Maritime Transportation Safety Information System (mTIS), made by the Korea Maritime Transportation Safety Authority (KOMSA), with the grid method, it needs to be verified legally before use. Thus, in this study, we validated the data from mTIS based on written judgment provided by the Ministry of Oceans and Fisheries (MOF) to perform a radiological risk assessment more appropriate for Korean situation. After that, an input variable for marine accident probability calculation "accident frequency" was quantified to examine the accident probability. We compared all 9 statistical data MOF offers and 3 matched the mTIS data. We verified that these three datasets are "statistical data" referenced by the STATISTICS ACT, confirming that mTIS accident data is legally valid and analyzed the Korean maritime accident frequency from 2017 to 2022. In terms of the frequency analysis, the criteria are vessel type, tonnage, sea area, accident type, vessel age, month, and season. We found that all the frequencies show a similar tendency for 6 years. Discovering the frequency of accidents is different by the distance from the coastline, we derived accident weights for each sea area in the East, West, and South Seas.

In this study, we verified the mTIS data to perform a risk assessment using a grid method more suitable for Korea, and developed an accident frequency database (DB) for each type and a sea area weight DB using mTIS. These DBs can be used as input variables for the radiological risk assessment model of maritime transport, which enables radiological risk to be quantified by applying probabilistic methods.

Acknowledgements

This work was supported by the Korea Foundation Of Nuclear Safety (KoFONS) grant funded by the Korean government's Nuclear Safety and Security Commission (NSSC) (No. 2106011).

Keywords: Radioactive waste, Maritime transport, Risk assessment, Probabilistic method

Experimental Study of Cellulose Degradation in the 1st Phase Domestic Repository and Its Estimated Impact

Hyun Woo Song^{1,*}, Moonoh Kim¹, Sang June Park², and Jun-gi Yeom²

¹KONES Corp.

²Korea Hydro & Nuclear Power (KHNP)

*shw@kones21.com

ISA generation from cellulosic waste degradation has been issued to enhance radionuclide release from the radwaste repository as ISA forms ligands with radionuclides, leading to increased solubility and decreased sorption. Dry active wastes, such as paper and clothes, generated from domestic nuclear power plants contain cellulose, thus raising the concerns of ISA generation. This study assesses the disposal safety of cellulosic waste under domestic 1st phase repository condition through literature review and experimentation.

According to literatures, ISA concentration greater than 1.00E-04 M would enhance the radionuclide release. To achieve such ISA concentration, disposal condition should maintain pH greater than 12.5 for cellulose degradation, and high Ca²⁺ concentration for high ISA yield rate, which are achievable in cementitious repository. However, the domestic 1st phase repository condition is estimated to maintain pH below 12.4, dissatisfying the cellulose degradation condition, so the cellulose degradation is unlikely to occur.

Experiment was conducted to investigate the disposal safety of cellulosic waste under domestic 1st phase repository. The experimental solution was conditioned to simulate the predominant ions (Na, Ca, K) of the pore water of the repository, and pH of the solution was set between 12.5~13.0, conservatively. Average of 30 mg/L of cellulose powder and 150 mL of the experimental solution were added to PTFE containers, then the samples were separately allocated into heating baths, each set at 95°C and 25°C.

After 3 months, the solution was filtered to remove remaining cellulose power, then mass change was measured after drying. From the result, the acceleration factor, derived via Arrhenius Equation, was calculated to be 11,682. Therefore, the experimental result would represent approximately 2,880 years after disposal. The ISA concentration of the solution was measured to be about 8.0E-06 M, which means even after the estimated design life of repository (1,645 years), the ISA concentration maintains below the threshold concentration (1.0E-04 M).

Considering the degree of polymerization (DP) of the cellulose powder used in the experiment was 350, actual cellulosic wastes with greater DP would takes longer time to degrade compared to the experimental result. Furthermore, like 1BLA of SFR, which was excluded from cellulose disposal safety assessment due to its pH evaluated to be about 12.4, the domestic repository also does not satisfy the cellulose degradation condition. Thus, disposal of cellulosic waste would not cause ISA issue within the domestic 1st phase repository. In the future, this study could be used as fundamental data for planning waste acceptance criteria.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. RS-2303-00234190).

Keywords: Cellulose degradation, ISA, Disposal environment, Safety assessment

KRID's Demonstration Plan for the Disposal of Spent Resin Waste Generated From Wolsong NPP PHWR

Seung Ho Lee*, Homyeong Lee, Junghyun Park, and Ji Hun Han

Korea Research Institute of Decommissioning (KRID)

*Sh4244@krid.re.kr

Recently, the efficient and cost-effective disposal of radioactive waste from nuclear power plants (NPPs) has gained attention due to the long-term operation of NPPs. The Korea Research Institute of Decommissioning (KRID) was established to complete the full life cycle of NPPs by conducting research on the back-end cycle, including cutting, decontamination, radioactive waste reduction, and radiochemistry of decommissioning waste. Among these efforts, KRID has been conducting a demonstration procedure for the volume reduction of intermediate-level radioactive waste, such as spent resins from Wolsong#1, a CANDU-type NPP. This initiative is based on years of fundamental technology development by SK T&S, KAERI, and UNIST. These institutions are developing equipment for spent resin disposal, which is expected to be completed by the end of 2024. KRID will collaborate with these institutions to conduct demonstration experiments starting in 2025.

In this presentation, we will outline the plans for the demonstration procedure of equipment designed to reduce and dispose of spent resin radioactive waste. We are constructing laboratories to support the demonstration, including a mock-up laboratory and a research building for multi-purpose R&D, both of which are expected to be completed by 2025. The research building will house the equipment for spent resin disposal, which will be operated as part of the demonstration. Additionally, we will discuss the demonstration plan for waste reduction and its verification.

Acknowledgements

This research project was carried out with support from the Korea Institute of Energy Technology Evaluation and Planning (Nuclear Power Plant Decommissioning Competitiveness Enhancement Technology Development Project funded by the Ministry of Trade, Industry and Energy (Establishment of radionuclide analysis system for below the low-level decommissioning wastes and the quality management system for the validation of radionuclide analysis procedure, RS-2023-00234176).

Keywords: Decommissioning, Radionuclide analysis, R&D project, KRID

Numerical Analysis on a Hollow Electrode Plasma Torch for Melting Application of Non-combustible Radioactive Wastes

Nam-Ki Lee, Dong-Hyun Lee, Sang-Yun Paek, Sun-Woo Kim, Hyo-Jeong Kim, and Jun-Ho Seo*

JeonBuk National University (JNBU)

*jhseo@jbnu.ac.kr

Normally, non-metallic wastes, such as sands, concrete and asbestos are regarded as electrically non-conductive materials. However, when the temperatures are increased up to the melting point, their electrical conductivities can be greatly improved, flowing arc current. Accordingly, these non-metallic wastes can be efficiently treated by heating them up to the electrically conducting temperatures by using a non-transferred type plasma torch, and then, melting them completely with arc currents in transferred mode of plasma torch. For this purpose, we propose a hollow electrode plasma torch consisting of three cylindrical electrodes (rear electrode, front electrode and exit nozzle). Compared with conventional plasma torch with two cylindrical electrodes (rear electrode and front electrode), the proposed plasma torch can provide more stable plasma jet specifically in reverse-polarity electrical connection at the condition of non-transferred mode due to the presence of the exit nozzle, resulting in rapid heating of the non-conductive materials. In this work, we investigate the basic performance of the proposed torch, such as voltage-current characteristics, thermal efficiencies and exit enthalpy distributions by solving MHD (Magneto-Hydrodynamic) equations numerically inside of the three electrodes. Detailed numerical results will be presented in poster session together with some experimental results.

Acknowledgements

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Keywords: Numerical analysis, Hollow electrode plasma torch, Non-conductive waste, Non-combustible waste

Necessity and Effectiveness of Introducing a Disposal Compatibility Review System to Ensure Acceptance of Low-Level Radioactive Waste

Jaechul Ha*, Jinwoo Jeong, and Kyunghyun Kwak

Korea Radioactive Waste Agency (KORAD)

*jcha@korad.or.kr

All low-level radioactive waste generated in Korea is scheduled to be disposed of at the Gyeongju Low-Level Radioactive Waste Disposal Facility, and the first phase of the caven disposal facility is currently in operation, and 200L drum-sized waste packages are being accepted and disposed of. It is necessary to consider the introduction of various types of radioactive waste and containers in the future.

Now that the Phase 1 disposal facility is operating stably, there is an increasing number of requests from generators for the acceptance of other types of radioactive waste besides the radioactive waste in standard containers. KORAD, as the only waste management company in Korea, must listen to the needs of generators and consider how to solve them. Based on this background, KORAD has derived a disposition suitability review system to resolve generators' grievances. From the generation of radioactive waste to its final disposal, a series of processes, explanations of essential requirements for each process, and review items to minimize gaps in terms of overall safety and suitability for disposal have been derived. In order for the system to be properly implemented, it is essential for the generator to provide information on the characteristics of the waste in the first step of the process, and it is also necessary for KORAD's relevant departments to conduct a close review of each major item at the same time. For the smooth application of this review system The interest and efforts of all stakeholders in waste management are required for the smooth application of this review system.

Keywords: Disposal facility, Disposition suitability review system, Aceeptance process

Study on Feasibility of Plasma Treatment Technology for Concrete Waste With Epoxy Paint Generated by NPPs

Sunghoon Hong*, Young Hwan Hwang, Jeongsu Jeong, Ji-Soo Yoon, Seong-Sik Shin, Jung-Kwon Son,
and Cheon-woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sunghoon.hong@khnp.co.kr

KHNP CRI has developed and is operating the plasma treatment facility to assess the applicability of the technology. In this study, demonstration test was carried out using the concrete with epoxy paint as a simulant waste. To process the concrete waste in plasma treatment facility, the additive for melt was developed using an electric box furnace and a slag prediction model. The simulant with additive was fed into the plasma melter through a vibrating feeding system with 8 kg/hour. The plasma torch operated in transferred mode, with power ranging between 94 to 101 kW. A total of 80 kg of waste was successfully treated, resulting in 56 kg of final product formed in the mold. During the experiment, the quantity of harmful substances in the off-gas released into the atmosphere was measured. Additionally, XRD (X-ray diffraction) and XRF (X-ray fluorescence) analyses were performed on the final product to evaluate its composition and properties.

Keywords: Plasma treatment, Concrete with epoxy paint, Demonstration test, Applicability

Development of Prototype of Handling Device With Radiation Shielding Applied for Large Concrete Drums in Kori Unit 1

Sunghoon Hong^{1,*}, Young Hwan Hwang¹, Seong-Sik Shin¹, Seok-Ju Hwang¹, Jeongsu Jeong¹, Jung-Kwon Son¹,
Cheon-woo Kim¹, Kwangsoo Park², and Changgyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*sunghoon.hong@khnp.co.kr

In the Kori unit 1 of radioactive waste storage facility, large concrete drums are stored for radioactive wastes such as concentrated waste, spent resin and spent filter. There are two types of drums: round-type and square-type drums, and require specialized handling due to their weight and size. The development of a handling device for these large drums is essential to safely manage the radioactive waste, minimize worker radiation exposure, and account for the limited space within the storage facility.

In this study, the materials and thickness selection of the radiation shielding were investigated.

Keywords: Handling device, Large concrete drum, Radiation shielding

Review on Sampling Methods for Selecting a Representative Drum of Large Concrete Drums in Kori Unit 1

Sunghoon Hong^{1,*}, Young Hwan Hwang¹, Seong-Sik Shin¹, Seok-Ju Hwang¹, Jeongsu Jeong¹, Jung-Kwon Son¹,
Cheon-woo Kim¹, Kwangsoo Park², and Changgyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*sunghoon.hong@khnp.co.kr

Large concrete drums are stored for a long time including concentrated waste in the Kori unit 1 of radioactive waste storage facility. The drums were produced before the waste acceptance criteria (WAC) was established. To dispose of the drum, it is essential to conduct a waste characterization of its internal radioactive waste, and to perform this characterization, sampling must be carried out. This study aims to review various sampling methods for selecting a representative drum.

Keywords: Large concrete drums, Sampling methods, Representative drum

A Study on the Melting Facility for the Treatment of Metallic Radioactive Waste

Taeseob Lim^{1,*}, Kwangsoo Park¹, Chungsik Lim¹, Changkyu Kim¹, HaeWoong Kim¹, Woonyun Choi¹,
Donghun Pak¹, Nari Kim¹, and Jung-Wook Cho²

¹Doosan Enerbility

²Pohang University of Science and Technology (POSTECH)

*taeseob2.lim@doosan.com

A total of 12 nuclear power plants in Korea are scheduled to reach the end of their design life by 2030 if they are not authorized to continue operating. If all of these plants are dismantled, radioactive waste will be generated in a short period of time. According to data provided by the IAEA, decommissioning of one PWR generates 650 tons of radioactive metals and 3,500 tons of contaminated metals, which is 67 percent of the total decommissioning waste. In addition to the metal waste generated during decommissioning, operating plants also generate large amounts of metal waste.

To reduce the volume of metallic radioactive waste and decontaminate, melting methods have been studied. During melting of metallic radioactive waste, nuclide elements are distributed in three forms: volatilization, slag, and metal residue. Nuclides such as Ag, Cs, Na, Se, Te, and Zn are volatilized at high temperatures and can be captured by bag filters in the dust collector. Nuclides such as Am, Ca, Ce, Cm, Eu, Nb, Np, Pa, Pm, Pu, Ru, Sb, Sm, Sr, Th, U, and Zr are oxidized and incorporated into slag, so they can be decontaminated by removing the slag that floats to the top of the metal melt. Nuclides such as Co, Fe, Mn, and Ni are not oxidized and remain in the metal, so it is not possible to decontaminate them immediately. The doses of these nuclides, except for Co-60, are not high, and Co-60 has a short half-life of 5.7 years, so the ingots can be stored in a radioactive waste repository until the doses are low enough to be processed.

To ensure the radiological safety of the workers, remote transporting devices are used to load the metal radioactive waste into the melting furnace. The dust collector, which is located above the crucible, for capture volatile nuclides and prevent fume dispersion, must be completely removed before the radioactive waste is loaded. The electric induction furnace uses a high-frequency oscillator to heat the furnace, and the melting temperature is between 1,600°C and 1,700°C.

Since the frequency of the oscillator, the required power of the cooling system, and the performance of the dust collection system are determined by the ingot production rate, this study suggests the appropriate facility devices for radwaste treatment.

Keywords: Metallic radioactive waste, Decontamination, Volume reduction, Melting facility, Radwaste treatment

Development of Immobilization Strategy for Efficient Treatment of Spent Filters

Moonoh Kim^{1,*}, Narae Lee¹, Jisoo Yoon², and Jun-Seok Nam³

¹KONES Corp.

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

³Korea Hydro & Nuclear Power (KHNP)

*mokim@kones21.com

To advance the decommissioning of Kori Unit 1 nuclear power plant, it is essential to treat the waste that remains unprocessed and stored. One of the unprocessed wastes is the spent filters stored temporarily in the filter room. These filters, numbering around 260, vary in radioactivity from intermediate to very low levels. Treating their disposal requires urgent development of appropriate treatment technologies.

Three primary treatment methods are under consideration: volume reduction through compression or shredding; shielded drum packaging for high-radioactivity waste; and a combined approach of volume reduction followed by shielded or non-shielded drum packaging. Regardless of the chosen method, if the concentration of long-lived radionuclides (with a half-life greater than 20 years) in any spent filter package exceeds 76,000 Bq/g, the packaging of spent filters must be immobilized.

However, if immobilization is to be applied only to spent filters that meet the immobilization requirements, the nuclide concentration of each spent filter must be identified and appropriately grouped and packaged to avoid exceeding the immobilized nuclide concentration limit.

For this selective immobilization, a complex process is required to find and retrieve spent filters that meet the conditions, which necessitates an assessment of whether selective immobilization can provide significant safety and economic benefits. Selective immobilization should be considered when applying it to spent filter treatment significantly improves operator safety and economic benefits, but otherwise, an efficient immobilization strategy should be established.

In this paper, we propose an efficient immobilization strategy for spent filters. We compare selective immobilization, where only filters meeting the immobilization requirements are processed, with a more generalized approach. Our goal is to present a strategy that minimizes worker exposure and economic costs while meeting regulatory standards. We evaluate both methods to determine the most effective approach for immobilizing spent filters in the context of Kori Unit 1's decommissioning.

Acknowledgements

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Keywords: Decommissioning, Spent filters, Immobilization, Treatment, Disposal

Comparative Analysis of the Economic Feasibility of PC-HIC and PE-HIC

Moonoh Kim*, Hyun Woo Song, and Narae Lee

KONES Corp.

*mokim@kones21.com

In Korea, approximately 890 polyethylene high integrity containers (PE-HIC) have been used to store spent resin for long periods of time. However, PE-HIC does not meet disposal suitability in terms of structural integrity and is therefore unable to be delivered to disposal facilities. To solve this problem, efforts are being made to develop overpacking containers that can compensate for the weaknesses of PE-HIC, but in the long term, it is essential to evaluate its competitiveness with rubber-component containers made of high-strength polymer concrete (PC-HIC).

PC-HIC is made of three materials - polyethylene, polymer concrete and carbon steel - and is available in two variants: 860 liter and 510 liter capacity. PC-HIC has been evaluated to satisfy structural and chemical integrity over the 300-year design life required for high-integrity containers (HIC), and has the advantage of not requiring a separate overpacking container. This durability makes PC-HIC an attractive option for long-term waste management.

To evaluate the competitiveness of PE-HIC and PC-HIC, various factors such as container robustness, ease of handling and manufacturing, and price must be considered. However, this study evaluated the competitiveness of the two HICs by focusing on their economic feasibility and manufacturing costs. Comparing the economic feasibility of the two containers clearly revealed the pros and cons of each.

While PC-HIC provides excellent long-term structural integrity, eliminating the need for overpacking, initial manufacturing costs can be high due to the materials used. On the other hand, PE-HIC has an economical initial cost, but it incurs additional costs due to overpacking and does not meet the structural soundness required for disposal. Therefore, to improve the economics of both containers, measures such as optimizing the manufacturing process, reducing material costs, or improving the design to improve structural performance should be considered.

Ultimately, economic assessments require a balanced approach that considers both initial and long-term costs and should aim for sustainable waste management solutions that meet safety and regulatory requirements. This study focus on developing solutions that improve the economic competitiveness of PE-HIC and PC-HIC while maintaining essential safety and structural standards for Intermediate and Low Level Waste disposal.

Acknowledgements

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Keywords: Glass fiber, Colloid, Radioactive waste, LILW disposal Facility, Safety assessment

~~The Suggestion for the Waste Acceptance Criteria of the Gyeongju Low- and Intermediate-Level Radioactive Waste Disposal Facility~~

~~Sang-June Park^{†,*}, Seongjin Maeng[†], Minji Park[†], Hyun-Woo Song², Moonoh Kim², and Jun-Gi Yeom[†]~~

~~[†]Korea Hydro & Nuclear Power (KHNP)~~

~~²KONES Corp.~~

~~*sjpark21@khnp.co.kr~~

Withdrawal

A Review of Glass Waste Form for Radioactive Iodine Immobilization

Seong-Sik Shin¹, Sung-Hoon Hong¹, Jung-Kwon Son¹, Cheon-Woo Kim¹, Young Hwan Hwang¹,
and Hyun Woo Kang^{2,*}

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Korea Atomic Energy Research Institute (KAERI)

*seongsik95@khnp.co.kr

Radioactive iodine, generated through nuclear fission in nuclear power plants, is highly soluble in water and exhibits high volatility. Radioactive iodine can pose serious risks to humans and the environment. Spent nuclear fuel primarily contains iodine isotopes, such as I-127 (stable), I-129 (half-life 1.57×10^7 years), and I-131 (half-life 8 days). Owing to the extremely long half-life, I-129 is the most abundant radioactive iodine isotope. Therefore, capturing and safely disposing I-129 during reprocessing and/or treating the spent nuclear fuel is essential. This paper reviews the properties of glass waste forms developed to immobilize radioactive iodine, focusing on different compositions and fabrication processes. It could serve as a foundation for future efforts in radioactive iodine immobilization.

Keywords: Fission product, Radioactive iodine, Glass waste form

Development FEP Scenarios and Impacts for the Safe Disposal of Concrete Drums

Seong-Sik Shin, Sung-Hoon Hong, Jung-Kwon Son, Cheon-Woo Kim, and Young Hwan Hwang*

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

In the 1980s and 1990s, prior to the introduction of the concentrated waste drying system (CWDS), Korea solidified boron concentrates using cement in 200 L drums, which are encapsulated in large concrete drums and currently stored in temporary storage facilities. To handle the concrete drums, Features, Events, processes (FEPs) were developed for processes such as concrete drum transportation, cutting and chipping of concrete drums, retrieval and transport of inner 200 L drums. In this study, representative scenarios were selected for FEPs that are expected to significantly impact safety, considering the probability of occurrence and potential safety effects. Safety assessment input data for these scenarios were also derived.

Keywords: Concrete drum, FEP scenario, Safety assessment

Development of FEP for the Safe Disposal of Concrete Drums

Seong-Sik Shin, Sung-Hoon Hong, Jung-Kwon Son, Cheon-Woo Kim, and Young Hwan Hwang*

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

During the 1980s and 1990s, prior to the introduction of the concentrated waste drying system (CWDS), Korea solidified boron concentrates using cement in 200 L drums, which are currently stored in temporary storage facilities. These drums are encapsulated in large concrete drums, and ensuring their disposal suitability is crucial for further treatment and disposal. The objective of developing Features, Events, Processes (FEP) is to identify potential risks and vulnerabilities in the process, as well as to suggest improvements that align with safety goals. In this study, FEPs were developed for processes such as concrete drum transportation, cutting and chipping of concrete drums, retrieval and transport of internal 200 L drums.

Keywords: Boron concentrated cement waste form, Concrete drum, FEP

Tellurite Glass for Radioactive Iodine Immobilization

Seong-Sik Shin¹, Sung-Hoon Hong¹, Jung-Kwon Son¹, Cheon-Woo Kim¹, Young Hwan Hwang¹,
and Hyun Woo Kang^{2,*}

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Korea Atomic Energy Research Institute (KAERI)

*hyunwookang@khnp.co.kr

The use of ^{235}U as a fuel in nuclear power plants results in the formation of various fission products. This study focuses on ^{129}I , a fission product and major iodine species among the iodine nuclides produced by nuclear fuel. ^{129}I has a long half-life of 15.7 million years, indicating the importance of its long-term disposal. Moreover, iodine is well-known to volatilize easily; when humans are exposed to radioactive iodine, it could accumulate in the thyroid gland and cause thyroid cancer. Moreover, when exposed to the environment, it severely impacts the ecosystem. Hence, radioactive iodine should be separated from the ecosystem. In this study, the tellurite glass system is proposed as an iodine immobilization matrix. Two type of glasses, Agl glass waste form and Cul glass waste form, were developed. Process and compositions for vitrifying iodine were developed, and chemical durability test (PCT-A) was conducted to confirm disposal suitability.

Keywords: Fission product, Radioactive iodine, Tellurite glass waste form

Review on Ultrasonic Imaging Methods for Understanding the Interior of Large Concrete Drums in Kori Unit 1

Sunghoon Hong^{1,*}, Young Hwan Hwang¹, Seong-Sik Shin¹, Ji-Soo Yoon¹, Jeongsu Jeong¹, Jung-Kwon Son¹,
Cheon-Woo Kim¹, Kwangsoo Park², and Changgyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*sunghoon.hong@khnp.co.kr

In the past, a large number of drums including concentrated waste, spent resin and spent filter have been repackaged into large concrete drums and stored in the Kori Unit 1 storage facilities. These drums contain waste generated before the establishment of the waste acceptance criteria. To dispose of the waste in the large concrete drums, it is necessary to conduct a disposal suitability assessment. To verify if the waste in the large concrete drums meets the disposal suitability criteria, it is essential to collect samples through coring and perform characterization. Therefore, methods need to be devised to facilitate the sampling process, considering the internal structure of the large concrete drums and the inner drums. This study reviewed various ultrasonic imaging technologies for understanding the internal structure of large concrete drums.

Keywords: Large concrete drums, Sampling process, Ultrasonic imaging technologies

Modeling the Pore Water Chemistry During Degradation of Cement Using the PHREEQC Geochemical Software

Myunggoo Kang* and Jaechul Ha

Korea Radioactive Waste Agency (KORAD)

*kkang09@korad.or.kr

Portland cement, used as a solidification, backfill, and construction material, is considered to determine the chemistry of a low and intermediate level radioactive waste repository for very long periods of time. The determination of a cement pore water composition and understanding its evolution with time underpins many radioactive waste disposal issues, such as buffer erosion, canister corrosion, and radionuclide solubility, sorption, and diffusion. Previous modelling approaches have tended to ignore the concrete degradation process, a consequence of which is that the change of water composition and cement properties can't be dynamically reflected in the repository system. Here, we investigate the applicability of the cement degradation model, that incorporates cement-water reaction as an integral component. Thermodynamics has been successfully applied to the field of cement hydration to predict phase assemblages and pore solution changes. This paper presents the steps required to analyze the degradation of engineered barriers by groundwater using PHREEQC geochemical software with Notepad++ editor. The analysis begins with determining which solid phases are thermodynamically predicted to precipitate and form using the oxide compositions of cements used the underground silo of Wolseong low and intermediate level waste (LILW) disposal facility in Korea. When the phases are known, PHREEQC is programmed to provide predictions of the phase dissolution and phase assemblage over time as well as the pore solution chemistry.

Keywords: Cement hydration, Cement degradation, Pore water chemistry, Geochemical software

Evaluation on Various Non-destructive Methods for Observing Inner Structure of Large Concrete Drums

Sunghoon Hong^{1,*}, Young Hwan Hwang¹, Seong-Sik Shin¹, Ji-Soo Yoon¹, Jeongsu Jeong¹, Jung-Kwon Son¹,
Cheon-Woo Kim¹, Kwangsoo Park², and Changgyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*sunghoon.hong@khnp.co.kr

The large concrete drums stored at Kori Unit 1 radioactive waste storage facility contain wastes generated before the establishment of WAC (waste acceptance criteria). To dispose of the internal waste in these large concrete drums, it is necessary to verify the internal structure and check for potential waste leakage due to defects or crack using non-destructive testing methods. This study reviewed various non-destructive techniques for assessing the structure and integrity of the large concrete drums, inner drums and their internal components.

Keywords: Large concrete drums, Non-destructive testing method, Internal structure

Characterization of Zr Added DG-2 Glass

Young Hwan Hwang*, Seong-Sik Shin, Sunghoon Hong, Seok-Ju Hwang, Jung-Kwon Son, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

Various radioactive wastes are formed during the operation of nuclear power plants. The dry active waste (DAW), boron concentrates, spent resin, and spent filters are primary streams. Vitrification attracts great attention for the solidification of various materials. It has been applied to radioactive waste to improve its disposal safety and decrease the volume. Vitrification comprises thermal decomposition of radioactive waste and formation of glass material. During the thermal decomposition step the primary organic species are decomposed. The KHNP developed the glass composition for the vitrification of DAWs.

In this paper, the effect of Zr in DG-2 glass is systematically studied. The DAW that contains metal particulate might affects the final composition of the glass. The characteristics of vitrified form is analyzed, including compressive strength, microstructure, product consistency test (PCT), leachability index, etc.

Keywords: Vitrification, DG-2, Zr, DAW

A Study on Re-Solidification of Boron Concentrates Using Geopolymer

Young Hwan Hwang*, Seong-Sik Shin, Sunghoon Hong, Ji-Soo Yoon, Jung-Kwon Son, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

During the operation of nuclear power plant, boron concentrates waste is generated from treatment of liquid waste treatment system. The boron concentrates have been treated using various techniques, such as cement solidification, paraffin solidification, high integrity container, etc. Currently, the boron concentrates are dried and stored on-site. The waste acceptance criteria (WAC) of KORAD defines that the boron concentrates should be solidified or packaged in high integrity container. In this study the geopolymer, which is one of the attractive materials for the solidification of radioactive waste, based boron concentrates solidification will be discussed. The characteristics of the geopolymer solidified radioactive waste also will be discussed.

Keywords: Geopolymer, Boron concentrates, WAC, Disposal

A Study on Pre-Treatment Equipment for Large Concrete Drums

Young Hwan Hwang^{1,*}, Seong-Sik Shin¹, Sunghoon Hong¹, Ji-Soo Yoon¹, Jung-Kwon Son¹, Cheon-Woo Kim¹,
Kwang-Soo Park², and Chang Kyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*yhhwang7@khnp.co.kr

Large concrete drums, which contains solidified boron concentrates or spent resin, are temporarily stored in nuclear power plant. Since some of inner drums contain relatively higher amount of radioactive nuclide compared to the conventional radioactive waste, they are re-packaged in large concrete drums for safe storage. The information of inner radioactive drums is not sufficiently obtained in this moment, the pre-treatment of them are required. The pre-treatment is comprised of segmentation of the outer drum, treatment of inner drums, etc.

In this paper, the segmentation process using wire saw of the large concrete drum will be discussed. The pilot test results using optimized condition and equipment will be demonstrated. Also, the local ventilation system for the reduction of secondary dust waste development status will be discussed.

Keywords: Large concrete drum, Segmentation, Disposal, Treatment

A Review on Management of Intermediate Level Waste and Interim Storage

Young Hwan Hwang*, Sunghoon Hong, Seok-Ju Hwang, Ji-Soo Yoon, Jung-Kwon Son, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

During the operation and decommissioning of nuclear power plant various level of radioactive waste are generated: high level waste, intermediate level waste (ILW), low level waste, and very low level waste. The intermediate level contains relatively high ratio of radioactive inventory. The careful management and control of them are important for the safe operation of nuclear power plant. The total amount of radioactive inventory of the disposal facility is strictly limited for the safe operation of it. Some ILW, such as spent resin, active metal components, primary circuit solid waste, etc., are temporarily stored on-site. In this paper, the international management of ILW will be discussed. In some cases, the temporarily stored ILW will be planned to disposed in geological repository. The current status and plan on the ILW management will be discussed in this paper.

Keywords: Intermediate level waste, Interim storage, ISFSI

A Study on Safety Analysis of Radiation Exposure During Treatment of Large Concrete Drums

Young Hwan Hwang¹, Seong-Sik Shin¹, Sunghoon Hong¹, Ji-Soo Yoon¹, Jung-Kwon Son¹, Cheon-Woo Kim¹,
Kwang-Soo Park², and Chang Kyu Kim²

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Doosan Enerbility

*yhhwang7@khnp.co.kr

The treatment of radioactive waste usually involves the radiation exposure to workers. The internal and external exposure will occur during the segmentation process of large concrete drums. The radiation exposure evaluation method will be discussed in this paper. Various factors are considered for reasonable evaluation of the radiation exposure. The practical scenario, including segmentation process, waste clearance process, etc., and resulting radiation exposure will be discussed in this paper.

Keywords: Radiation exposure, Large concrete drums, Radioactive waste, Waste clearance

A Study on Treatment Technology for Repackaged Large Concrete Drums

Jisoo Yoon*, Young Hwan Hwang, Seong Hoon Hong, Seong Sik Shin, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*jisooYoon1@khnp.co.kr

For untreated radioactive waste generated during the operation of Kori Unit 1 and stored in the NPP for a long time, it is required to secure the suitability of disposal for timely treatment. Large concrete containers generated by concentrated waste liquid and repackaging waste resin generated in the past were generated before the establishment of the waste acceptance criteria, and the suitability of disposal has not been confirmed, so it cannot be disposed of as it is now. In order to treat the concrete drum embedded waste to ensure the suitability of disposal, it is necessary to check the current expected condition of the embedded waste and perform an appropriate treatment process to satisfy the acquisition standard according to the condition of the embedded waste. In this study, the treatment technology of waste resin was investigated, and the main technologies are free water removal technology and solidification treatment technology.

The free water removal technology in the waste resin refers to a technology that creates a disposition-suitable state by removing free water from the slurry-state waste resin, and the volume may be reduced through this. However, there is also a method of reducing the volume of the waste resin itself by chemical treatment and incineration. The applicability of the free water removal technology through dehydration, centrifugation, drying, incineration, and chemical treatment was examined.

In addition, dehydration, drying, and volume-sensitive waste resin can be solidified or packaged using HIC if additional treatment is required for disposal compatibility pictorials. Solidification technology is confirmed to have been used in various countries such as the United States and Sweden, and research on various solidification media is being conducted, but research using cement, polymers, and geopolymers is the most active.

Keywords: Large concrete containers waste, Spent resin, Waste treatment technology

~~A Study on the Methods for Conditioning and Packaging of Spent Resin-
Overseas NPPs~~

~~Jisoo Yoon*, Young Hwan Hwang, Seong Hoon Hon, and Jung Kwon Son
Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)
*jisoo.yoon1@khnp.co.kr~~

Withdrawal

The Radiation Dose Assessment of the High-Temperature Catalytic Oxidation System for the Treatment of Spent Scintillation Liquid

Jun-Seok Nam^{1,*}, Mikyoung Yang¹, Suil Bang¹, Young-Suk Oh¹, Sung Jun Hong², Keon Hwa Lee²,

Woo Cheol Choi³, Jung Ho Song³, and Tae Hoon Jeon³

¹Korea Hydro & Nuclear Power (KHNP)

²Hana NPE Co.

³BRNC Inc.

*junseoknam@khnp.co.kr

The presence of organic components in spent scintillation liquid should be considered during all steps of radioactive waste processing for final disposal. Scintillation liquids often referred to as cocktails are generated from radiochemical analyses of radionuclides, which mainly consists of mixtures of liquid organic materials such as toluene and xylene. Typical features of these liquid organic materials are volatility, combustibility and toxicity. These are the reason why special attention must be paid to the management of liquid organic radioactive wastes. To select an appropriate waste management strategy and to design the treatment process of spent scintillation cocktails, authors investigated the specific radioactivity and moisture content of spent scintillation liquid generated at Wolsong nuclear power plants in the previous study. Among the various available technologies, the high-temperature catalytic oxidation technology was selected for the treatment of spent scintillation liquid in this work. The radiation dose impacts of liquid and gaseous radioactive effluent releases from the high-temperature catalytic oxidation system are evaluated using NRC Dose3. To examine the appropriateness of the calculation model, the radiation dose assessment will be performed using K-Dose60 and NRC Dose3 respectively and the results will be discussed.

Keywords: Scintillation liquid, Scintillation cocktail, Organic radioactive wastes, Liquid scintillation counter, Liquid scintillator

New Insights Into the Influence of Calcium on the Immobilization of Radioactive Borate Waste in Metakaolin-based Geopolymer Waste Forms

Byoungkwan Kim^{1,2}, Brant Walkley², John L Provis³, Hyun-min Ma¹, and Wooyong Um^{1,*}

¹Pohang University of Science and Technology (POSTECH)

²The University of Sheffield

³Paul Scherrer Institut

*wooyongum@postech.ac.kr

Alkali-activated geopolymers are promising materials for immobilizing radioactive borate waste (BW). However, the effectiveness of this immobilization still needs additional improvement due to the negative effects of borate on the physicochemical properties of geopolymers. In this study, the influence of $\text{Ca}(\text{OH})_2$ on geopolymer waste forms containing BW was investigated by varying the Ca/Al ratios (0.25–1.0). The findings revealed that $\text{Ca}(\text{OH})_2$ substantially improved the compressive strength of the geopolymers, especially at a Ca/Al ratio of 0.25, where the strength nearly tripled (from 5 to 14 MPa). This strength evolution was related to the promotion of geopolymerization and the formation of more C-(A)-S-H gel phase. While lower Ca/Al ratios did not significantly alter the geopolymer structure, higher ratios led to the predominance of C-(A)-S-H gel, as evidenced by solid-state ^{29}Si MAS NMR spectroscopy. Additionally, a range of analyses, including spectroscopic, thermal, and microstructural techniques such as thermogravimetry, in-situ high-temperature XRD, and SEM/EDS, showed that $\text{Ca}(\text{OH})_2$ contributed to the development of extensive geopolymer gels and a denser microstructure. Importantly, the amount of leachable B was reduced by more than half in the geopolymer prepared at a Ca/Al ratio of 0.25, which also exhibited the highest compressive strength. The leachability index of B increased from 7.5 to 8.7. As a results, the introduction of $\text{Ca}(\text{OH})_2$ increases the pH, which dissolves the physically immobilized BW and reduces the chemically immobilized BW by consuming reactive silicate. This study highlights the importance of optimizing Ca/Al ratios to improve the immobilization performance of geopolymers for containing radioactive BW.

Acknowledgements

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Keywords: Borate waste, Metakaolin-based geopolymer, Waste form, C-(A)-S-H, Immobilization

Development of Ceramic-Metal Waste Form for Immobilization of Radioactive Cesium

Ga-Yeong Kim^{1,2,*}, Ki-Rak Lee², Jung-Hoon Choi², Jae-Hwan Yang¹, and Hwan-Seo Park²

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*kky@kaeri.re.kr

Among the various radionuclides present in spent nuclear fuel, cesium is a heat-generating radionuclide, alongside strontium. Cesium and its decay product, barium, contribute significantly to the initial heat generation of spent nuclear fuel, necessitating that the immobilization matrix for cesium exhibits high thermal stability. Traditional matrices for cesium immobilization, such as glass and cement, suffer from limited chemical durability and thermal stability. To address these issues, ceramic matrices have been developed, with Cs pollucite being a notable glass-ceramic material known for its stable cesium immobilization properties. However, ceramic matrices pose a risk of significant temperature increase at the center of the waste form due to low thermal conductivity when subjected to high waste loading, potentially compromising the integrity of the immobilized waste and disposal environment. To mitigate these issues, Ceramic-metal (Cermet) materials, which combine the advantages of ceramics and metals, have been developed. Among the types of Cermet, metal-matrix composites (MMC) feature a metallic matrix with encapsulated ceramic phases. MMC waste forms offer both the high chemical durability of ceramics and the excellent thermal conductivity of metals.

This study aims to determine the optimal fabrication conditions for Cermet for cesium immobilization and to evaluate its characteristics. The target radionuclide, cesium, was fixed in Cs pollucite phases and encapsulated in various metallic matrices to ensure chemical durability and thermal conductivity. Precursors of the Cs pollucite waste form developed in previous studies were mixed with three types of metal powders at ratios of 80, 85, and 90wt%. The metal powders used were copper, SUS, and bronze (copper-tin alloy). The mixed powders were cold pressed using a hydraulic press and subsequently sintered. The green pellets were sintered in a tube furnace under a 3% H₂ atmosphere (2 L/min) at temperatures below the melting points of the metals for 5 hours. The sintering temperatures were set at 850°C for Cu and bronze, and 1,000°C for SUS. The characteristics of the manufactured Cermet were evaluated using X-ray diffraction, scanning electron microscopy, energy dispersive spectroscopy, differential scanning calorimetry, and Product Consistency Test Method A.

Keywords: Ceramic-metal, Waste form, Cesium, Immobilization, Sintering

Vitrification of Cement Solidified Waste Form Containing Boron Concentrate

Seong-Sik Shin, Sung Hoon Hong, Jung-Kwon Son, Cheon-Woo Kim, and Young Hwan Hwang*

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

In Korea, boron concentrates generated from the operation of pressurized water reactors (PWR) account for a significant portion of radioactive waste. In the 1980s and 1990s, before the introduction of the concentrated waste drying system (CWDS), Korea had solidified boron concentrates using cement in 200 L drums, and these are currently stored in temporary storage facilities. To dispose of radioactive waste, the solidified waste form must meet the waste acceptance criteria (WAC) of the Korean radioactive waste agency (KORAD) disposal facility.

In this study, to ensure the disposal suitability of boron concentrated cement waste generated in the 1980s and 1990s, a vitrification process was adopted. The simulated cement waste form was fabricated and analyzed, and additive glass composition, mixing ratios, and vitrification processes were developed.

Keywords: Boron concentrate, Waste acceptance criteria, Vitrification

5분과

제염해체 (ORAL)

Geopolymer-Based Encapsulation: A Study on Uranium Immobilization

Sajid Iqbal and Jong-Il Yun*

Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

Uranium (U) contamination in groundwater and soil represents a significant global threat to the environment and public health due to its radiotoxicity and chemical hazards. Large amounts of U can be introduced into ecosystems through nuclear accidents, mining activities, and leaks from waste generated by the Plutonium Uranium Redox Extraction (PUREX) process. In oxidizing conditions, U exists primarily as the mobile and soluble uranyl ion (UO_2^{2+}), which can readily migrate through water and soil via adsorption and ion exchange processes. Despite advances in immobilizing uranium using materials such as cement, bitumen, and glass, a sustainable long-term solution remains elusive.

Geopolymers have recently emerged as a promising alternative for encapsulating various radioactive wastes, including uranium, due to their low carbon footprint and resistance to alkali attack. Understanding uranium's behavior within geopolymer matrices is essential to optimize their effectiveness in immobilization and ensure the long-term stability of radioactive waste in deep geological repositories.

This study explores the encapsulation of uranium species in geopolymer matrices at concentrations up to 9wt%. The synthesized waste forms demonstrated excellent chemical durability, with leaching resistance as low as 10^{-6} g/m²/day. XRD analysis revealed a mixed-phase system, while XPS and XANES spectra confirmed that U(VI) is the predominant species. TGA results indicated good thermal stability up to 1,000°C. The presence of nitrate ions, forming sodium nitrates, appears to affect the geo-polymerization process and mechanical durability. Despite crack formation at higher U loadings, leaching resistance surprisingly increased, suggesting strong sorption between the negatively charged geopolymer surface and uranium, or the formation of complex uranium species. These findings underscore the potential of geopolymer materials for effective uranium immobilization.

Acknowledgements

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Keywords: Uranium, Solidification, Geopolymers, Radioactive waste management

Underpinning of Corrosion Behavior of Carbon Steel SA106 Gr.B Between HyBRID and Oxalic Acid

Kamal Asghar^{1,2}, Miguta Faustine Ngulimi^{1,2}, Sion Kim^{1,2}, Bum Kyoung Seo², and Changhyun Roh^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*chroh@kaeri.re.kr

Chemical decontamination processes have been used when operation and decommissioning of the nuclear power plants to reduce radiation exposure but the chemicals used may cause a back-corrosion of structural materials of basal metals. In this study, we elucidated comparative corrosion behaviors of carbon steel SA106 Gr.B (CS) in two different chemicals from Hydrazine-Based Reductive Ion Decontamination (HyBRID) and Oxalic Acid (OA) solution. Both chemicals have distinct mechanisms of decontamination and corrosion. We found that OA based chemical was less corrosive to CS compared to the HyBRID based chemical. Different conditions of HyBRID based chemical were used with and without the addition of copper to confirm the reductive effect, observing a higher corrosion rate with the addition of Cu ion. In contrast, the CS dissolved in OA based chemical has intergranular attack (IGA) across its surface. The corrosion rate of CS in different concentration of OA and HyBRID chemicals were found to be in the range of 0.52-0.71 $\mu\text{m/h}$ and 3.43-3.58 $\mu\text{m/h}$, respectively. Finally, we elucidated corrosive behavior of CS of basal metal between HyBRID and OA from reductive and chemical decontamination process.

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Keywords: Chemical decontamination, Corrosion, Carbon steel, HyBRID, Oxalic acid

Chemical Decontamination Process in a Nuclear Facility

Miguta Faustine Ngulimi^{1,2}, Sion Kim^{1,2}, Kamal Asghar^{1,2}, Bum Kyoung Seo², and Changhyun Roh^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*chroh@kaeri.re.kr

Nuclear energy accounts for approximately 10% of global energy production, positioning it as a promising solution for achieving carbon neutrality in the context of escalating climate change concerns. Nonetheless, the effective management of radioactive waste, which can remain hazardous for up to one hundred thousand years, presents considerable challenges that must be addressed to uphold public trust and safeguard environmental safety. In NPPs, contamination arises from neutron bombardment of corrosion products, creating activated radionuclides like Co-60, Co-58, Fe-59, Cr-51, and Mn-54. Coolant water corrodes metal pipes during operation, releasing metal ions that migrate to the reactor core. These ions undergo neutron activation and deposit with oxide layers on metal surfaces, leading to significant contamination issues. The accumulation of radioactively activated corrosion products within the oxide layer formed on the internal surfaces of the pipes of the primary coolant system of the nuclear power plants (NPPs) during operation significantly increases radiation exposure to workers during decommissioning. To mitigate this exposure and prevent the spread of radionuclides, chemical decontamination processes are employed, primarily targeting the dissolution of the oxide layer, which results in the release of bound radionuclides into a soluble form that can be removed. Several organic chemical decontamination processes, such as CORD (Chemical Oxidation Reduction Decontamination), CAN-DECON, and CITROX, have been widely used in the industry. While these methods are effective in reducing surface contamination, they are associated with drawbacks, including potential corrosion of the piping system and the generation of significant amounts of organic complex waste. In contrast, inorganic chemical decontamination methods, such as the Hydrazine-Based Reductive Metal Ion Decontamination (HyBRID) process proposed by the Korea Atomic Energy Research Institute (KAERI), offer an alternative approach. In this study, we introduced various and potent chemical decontamination processes.

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Keywords: Chemical decontamination, Primary cooling system, Oxide layer, Nuclear Facility

Multimodal Approach for Distribution Coefficient Prediction and Input Feature Importance Analysis

Seongyeon Na¹, Heewon Jeong², Ilgook Kim³, In-Ho Yoon^{3,*}, and Kyung Hwa Cho^{2,*}

¹Ulsan National Institute of Science and Technology (UNIST)

²Korea University

³Korea Atomic Energy Research Institute (KAERI)

*ihyoon@kaeri.re.kr, khcho80@korea.ac.kr

The nuclear power plant decommissioning market is growing worldwide, but radionuclide leakage remains a significant risk during the decommissioning process due to unforeseen events. Radionuclides entering the soil profile can remain for long periods by adsorbing the soil particles or migrate into groundwater via water movement. Sorption is a crucial mechanism governing radionuclide mobility in soil. The distribution coefficient (K_d) is used to quantify radionuclide adsorption by representing the ratio between the solid and liquid phases at equilibrium. Based on the adsorption concepts, batch soil experiments and isothermal adsorption models face limitations in experimental design for all variables and applicability on large scales. Similarly, previous machine learning studies in predicting K_d have constraints, as they typically overlooked the inherent properties of the soil.

We propose a multimodal algorithm for K_d prediction that fuses heterogeneous data across environmental domains: soil adsorption factors, X-ray fluorescence (XRF) data, and X-ray diffraction (XRD) spectra. The multimodal model achieved robust predictive performance by applying specialized sub-networks for each input data, with a coefficient of determination (R^2) of 0.87 and root mean squared error (RMSE) of 0.89 for natural log-transformed K_d values.

We utilized explainable artificial intelligence (XAI) techniques to address the interpretability challenges posed by the black-box nature of the machine learning model. Shapley Additive exPlanation (SHAP) analysis showed that pH and initial radionuclide concentration from the soil adsorption factors, and CaO from the XRF data, were key contributors to the model performance. Moreover, Grad-CAM analysis revealed that the model assigned significant weight to specific regions on the XRD spectra, focusing on meaningful adsorption spectral peaks rather than relying on high-intensity peaks. The XAI techniques used in this study are expected to assist decision-makers regarding soil remediation in decommissioning nuclear facilities by providing the prioritization of input variables used in K_d prediction.

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Keywords: Adsorption, Distribution coefficient (K_d), Multimodal model, Radionuclide, XRD spectrum

Investigating the Solubility of Metal/Metalloid Oxides in Choline Chloride-Based Deep Eutectic Solvents for Selective Recycling of Spent Uranium Catalysts

Kwangseo Kim, Stuart Aberdeen, Woohyun Lim, Shuang Liu, Richard I. Foster, and Sungyeol Choi*

Seoul National University (SNU)

*chosys7@snu.ac.kr

Uranium catalysts, used in various industrial processes like acrylonitrile production and hydrocarbon synthesis, consist of uranium mixed metal & metalloid oxides (UMOX, where M includes Sb, Fe, Te, Cu, W and Mo) impregnated onto inorganic SiO₂ supports (approx. 70 vol %) improving their chemical stability and catalytic ability. In South Korean textile company accumulated approximately 7,100 tonnes of these catalysts, classified as radioactive waste. Spent Uranium catalyst pose environmental risk if landfilled, including groundwater contamination. Therefore, the treatment and volume reduction of the spent catalysts through the separation of the metal & metalloids from the bulk SiO₂ support is desirable. Herein, it is important to develop a green and economic technology to extract and recycle hazardous metals from spent uranium catalysts.

Deep eutectic solvents (DESs), which combine a hydrogen bond donor (HBD) and acceptor (HBA), are recognized for their low melting points, cheap, and eco-friendliness. These solvents have been widely reported for their ability to dissolve metal oxides. Choline chloride (ChCl), a cost-effective HBA, is widely used in DES synthesis. This study investigated the solubility of a uranium-free spent catalyst (UFSC), with the same composition as spent uranium catalysts but no uranium, along with the key metal and metalloid oxides such as Sb₂O₃, SiO₂, and Fe₂O₃, using different types of choline chloride-based DESs. Additionally, the effect of hydration on oxide solubility was examined by varying the water content (5%, 10%, 20%, 30%, 40%, and 60%) in choline chloride:p-toluenesulfonic acid (ChCl:PtsA) mixtures. The UFSC was characterized before and after dissolution using scanning electron microscopy/energy-dispersive X-ray spectroscopy (SEM/EDS), X-ray fluorescence (XRF), inductively coupled plasma digestion (ICP digestion), and X-ray diffraction (XRD).

Both ChCl:Ethylene glycol and UREA DES exhibited poor solubility for both individual oxides and UFSC, while ChCl:PtsA dissolved each of the individual oxides well, but also struggled to dissolve the UFSC. This highlights the crucial role of the HBD in oxide dissolution and the importance of careful HBD selection in DES screening. Although the addition of water to ChCl:PtsA improved solubility for certain oxides, such as Sb₂O₅, the hydration effect on UFSC was insignificant. Consequently, further studies should explore methods including increasing temperature, adjusting agitation, modifying the solid-to-liquid ratio, or even advanced dissolution techniques such as electrochemistry and microwave assisted dissolution to enhance UFSC solubility. This study highlights the potential of DESs for recycling spent uranium catalysts and the selective extraction of metals, though optimization is necessary for effective UFSC dissolution.

Acknowledgements

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Keywords: Decontamination, Deep eutectic solvents, Metal/metalloid oxides dissolutions, Spent catalyst

New Reference Materials for Quality Assurance/Quality Control in Radioactive Waste Analysis for Nuclear Power Plant Decommissioning

Minju Lee, Yoonhee Jung, and Sang-Han Lee*

Korea Research Institute of Standards and Science (KRISS)

*s.lee@kriss.re.kr

The decommissioning of nuclear facilities has become one of the most significant projects in the global nuclear industry. In Korea, starting with the permanent shutdown of the Kori-1 nuclear power plant (NPP), the decommissioning of NPPs that have reached the end of their licensed operating life is expected. A large amount of radioactive waste, including concrete, soil, and metal, is generated during the processes of facility decontamination and dismantling. Accurate and precise radioactivity measurement plays a crucial role in waste classification, disposal, and cost reduction. ISO/IEC 17025 helps laboratories develop and maintain a quality management system that is adaptable and efficient in ensuring the validity of results. Among the quality assurance methods, it is generally recommended to use reference materials (RMs) or quality control materials, conduct intralaboratory comparisons, and test blind samples. The Korea Research Institute of Standards and Science (KRISS) has developed concrete, soil, and metal liquid RMs for measuring gamma-emitting radionuclides (^{241}Am , ^{134}Cs , ^{137}Cs , ^{60}Co , ^{152}Eu , and ^{154}Eu), beta-emitting radionuclides (^{90}Sr , ^{55}Fe , and ^{59}Ni), and alpha-emitting radionuclides (^{238}Pu and $^{239+240}\text{Pu}$) in radioactive waste. The new RMs were manufactured in accordance with ISO 17034. The homogeneity and stability of the RMs were confirmed to meet ISO Guide 35. The RMs have the potential to be used as proficiency test materials and for the verification of analytical methods in radioactive waste analysis. The RM manufacturing method employed in this study opens up various application possibilities, enabling the production of accurate and reliable RMs for different waste matrices.

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Keywords: Nuclear power plants, Decommissioning, Radioactivity measurement, Reference material, Concrete, Soil, Metal

Site-Specific Optimization of Radionuclide Remediation Strategies

In-Ho Yoon^{1,*}, Ilgook Kim¹, Youngho Sihm¹, Geun-Ho Kim¹, KwangSeong Jeong¹, Chan Woo Park¹,

Jun-Young Ahn², Seong-Chun Jun², and Jong-Hwa Yi²

¹Korea Atomic Energy Research Institute (KAERI)

²GeoGreen21 Co. Ltd.

*ihyoon@kaeri.re.kr

The decommissioning of nuclear facilities presents significant challenges in environmental remediation, particularly concerning radionuclide contamination in soil. This study introduces a novel methodology for developing optimal remediation strategies tailored to specific nuclear facility sites. Our approach integrates comprehensive soil characterization, radionuclide behavior analysis, and site-specific geological modeling to enhance the efficiency and effectiveness of remediation efforts. We conducted extensive investigations on soil samples from three distinct nuclear facility sites in Korea. Physicochemical properties including surface area, cation exchange capacity (CEC), and organic matter content were analyzed. X-ray fluorescence (XRF) and X-ray diffraction (XRD) techniques were employed to determine elemental composition and mineralogical properties. These analyses revealed significant variations in soil characteristics across sites, highlighting the necessity for site-specific remediation approaches. Desorption tests were performed using deionized water and nitric acid solutions of varying concentrations to evaluate the mobility of key radionuclides (Cs, Sr, and Co). Sequential extraction tests further elucidated the binding mechanisms of these radionuclides to different soil fractions. Results demonstrated distinct mobility patterns: Sr exhibited the highest mobility, while Cs and Co showed stronger retention in soil matrices. To simulate real-world conditions, we conducted column tests using contaminated soil layers. These tests corroborated the findings from batch experiments, showing differential vertical migration of radionuclides through soil profiles. The data obtained were crucial in developing a conceptual site model (CSM) that incorporates stratigraphic and permeability information. Based on our findings, we propose a decision-making framework for selecting appropriate remediation techniques. This integrated approach, combining detailed soil characterization, radionuclide behavior analysis, and site-specific modeling, provides a robust foundation for optimizing remediation strategies. By tailoring remediation efforts to site-specific conditions and contaminant behavior, this methodology promises to enhance the efficiency and cost-effectiveness of nuclear site decontamination processes.

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Keywords: Site-specific, Optimization, Radionuclide, Remediation strategies, Integrating, Soil characteristics, Contaminant behavior

5분과

제염해체 (POSTER)

Laser Cutting Study for Application to Dismantling of Calandria Shells in Heavy Water Reactors

Jae Sung Shin^{1,2}

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

jsshin12@kaeri.re.kr

Calandria shells in heavy water reactors are 28.6 mm thick, with an internal diameter of 7.6 m and a length of 4.0 m, and are made of 304 L stainless steel. Laser cutting is highly effective for dismantling calandria shells because it can cut thick metals at high speeds. Additionally, laser cutting generates very little secondary waste due to its narrow kerf width, low reaction force, and the ability to operate with only a small head due to fiber optic delivery.

For this purpose, this study was conducted to cut 30 mm thick stainless steel using a laser power of 6 kW. First, cutting performance was measured based on the focal length of the focusing lens to select a head suitable for practical application. Additionally, an appropriated cutting speed was determined by measuring the maximum cutting speed. Moreover, the amount of secondary emissions generated during cutting was also evaluated. Furthermore, a change-of-direction test and a long-length cutting test were performed to evaluate cutting stability under given process conditions.

Using the selected head, 30 mm thick stainless steel was cut at a speed of 270 mm/min. The average cutting speed was 1.18 mm and the amount of secondary emissions was 279.5 g/m. No special handling was required for the change-of-direction and long-length cuts, and all cuts were successful. These results are expected to provide useful data for future applications in calandria dismantling sites during the decommissioning of heavy water reactors.

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Keywords: Laser cutting, Laser processing, Calandria shell, Heavy water reactor, Nuclear decommissioning, Nuclear dismantling

Numerical Analysis on Melting Process of Non-combustible Wastes by a Transferred Arc Plasma

Jun-Ho Seo*, Shi-Young Yang, Hyo-Jeong Kim, Dong-Hyun Lee, Nam-Gi Lee, Son-Woo Kim,
and Sang-Yun Baek
JeonBuk National University (JNBU)
*jhseo@jbnu.ac.kr

Recently, plasma melting system is promising for volume reduction of non-combustible radioactive wastes generated during the decommissioning of nuclear power plants. Theoretically, plasma melting system can provide a unified melting process for both of metallic and non-metallic wastes, and the melted slugs can be produced in non-leachable glass (or slag) form, which are preferable in LILW (Low and Intermediate Level Waste) disposal facility. In addition, plasma melting route is expected to produce excellent volume reduction ratio compared with other technologies such as super-compaction. Accordingly, it is very important to understand melting process of non-combustible radioactive wastes by a plasma system for its practical application to the volume reduction of non-combustible radioactive wastes. However, plasma melting process, in particular using a transferred arc plasma, is a very complicated process including not only heat transfer but also phase transformations and magneto-hydrodynamics, requiring for high computational cost, and consequently, making it difficult to design a plasma melting system for radioactive waste treatment based on theoretical understanding on the melting process.

In this work, to alleviate this computational problem and provide basic information on the melting process by a transferred arc plasma, we investigate the melting process by a transferred arc plasma numerically, only accounting for heat transfer in the anode that was assumed as a non-combustible conducting waste. By solving transient heat equation including two phases (solid and liquid) in the anode region, pool profile was calculated for a given input current and arc efficiency. Then, from the calculation results, a practical melting time was predicted for simulated wastes consisting of 50wt% sand and 50wt% iron.

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Keywords: Numerical analysis, Plasma melting, Transferred arc plasma, Phase transformation, Non-combustible wastes

Performance Evaluation of Ion-Exchange Resins Under High-Pressure, Low-Temperature Conditions for Innovative Small Modular Reactor (i-SMR) Systems

Ukjin Yeo, Stuart Aberdeen, Richard I. Foster, and Sungyeol Choi*

Seoul National University (SNU)

*chosys7@snu.ac.kr

Ion-exchange (IX) will play a crucial role in the removal of ionic impurities in innovative small modular reactor (i-SMR) systems currently under development, particularly within the Modular Make-up and Purification System (MMPS). These impurities can form deposits on critical system components or increase corrosiveness, which can undermine the long-term operational stability of the reactor. Therefore, effectively removing these impurities is essential to ensuring the safety and efficiency of the reactor.

These reactors operate under high-pressure and low-temperature conditions, so the primary objective is to evaluate whether the IX resin can function stably and efficiently in a high-pressure environment. Therefore, we will assess the resin's durability against various physical and chemical stresses and examine its applicability in actual high-pressure systems.

In this study, preliminary experiments were conducted to evaluate the performance of nuclear-grade resins under high-pressure, low-temperature conditions, focusing on pH screening and isotherm adsorption experiments. For these experiments, Purolite™ NRW 150 and NRW 160 resins were used, for the removal of target ions: Co, Ni, Fe, Sr, and Cs. The pH screening experiments were conducted over a pH range of 1 to 5. The results showed that the resins exhibited optimal performance within a specific pH range, with high removal efficiency of the target ions. This suggests that maintaining the reactor environment within this specific pH range is crucial for maximizing ion removal efficiency.

Isotherm adsorption experiments were performed to determine the adsorption capacity of the resins for specific ions at various concentrations. The initial ion concentrations used for the experiments were 100, 500, 1,000, 2,000, 5,000, and 10,000 mg L⁻¹. The results indicated that the adsorption behavior followed the Langmuir isotherm model, showing monolayer adsorption on a homogeneous surface. These findings suggest that the selected IX resins are well-suited for the selective removal of target ions under the tested conditions.

The preliminary results will serve as a foundational basis for verifying the feasibility of resin utilization under iSMR operating conditions. The primary objective moving forward is to assess resin performance in high-pressure environments (150 bar), which are critical to i-SMR operations. Future experiments will focus on conducting detailed adsorption studies using a variety of cation and anion exchange resins under high-pressure conditions. These studies are designed to identify the most suitable resins for practical deployment. The combined data will play a crucial role in refining and selecting potential IX resins for this process.

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Keywords: Innovative small modular reactor, Ion-exchange resins, High-pressure conditions, pH screening, Isotherm adsorption

Review of Dose Assessment and DCGL Calculation Process for Decommissioning Nuclear Facilities

Hyung-Woo Seo*, Young-Il Na, Chan-Geun Park, and Gang-Woo Ryu

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sehywo1727@khnp.co.kr

The decommissioning nuclear power plants is defined as activities that can be excluded from the regulation of the Nuclear Safety Act on sites and remaining buildings after systems, structures, and components are removed. In the site survey stage, it is important to obtain the sampling results of the actual residual contamination concentrations and the DCGL values, which is a criteria for determining that its level is within the allowable range. Many studies have been conducted previously to derive DCGLs, and the results actually vary depending on the site characteristics of each nuclear power plant. In addition, for DCGL calculation, the RESRAD computer code are used in decommissioning nuclear facilities worldwide, and are also used in some other media (i.e. underground structures, buried pipes, etc.) Therefore, the purpose of this study is to review the license termination framework and guidelines for decommissioning nuclear power plants, including site characteristics data, for DCGL calculation, and also reviewed procedures to be considered when calculating DCGLs.

We reviewed the License Termination Plan (LTP) on decommissioning nuclear power plants in the U.S. for data related to DCGL calculation to confirm the aspects applied to the actual site at the license termination stage of the facility. Mainly referenced nuclear power plants include Zion, Rancho Seco, Connecticut Yankee, and Humboldt Bay, and the contents considered in the facilities were reviewed for sites and remaining buildings that are representatively considered release media. In addition, we referred to the contents of the NUREG-1757 on DCGL and flows of license termination at the decommissioning sites. The contents reviewed included media, scenarios, exposure pathways, etc. that should be considered when calculating DCGLs. We also confirmed that this was a flow that will be connected to the final stage of decommissioning, the Final Site Survey (FSS).

This study reviewed actual cases of overseas nuclear power plants to derive DCGLs in the final stage of nuclear decommissioning, and confirmed the points they considered for this purpose. In addition, the guidelines presented by NUREG-1757 were reviewed, and in particular, we looked into what items were used to derive DCGLs and tried to identify procedural aspects to organize them. In summary, this study reviewed the points to be considered for DCGL derivation in the final stage of nuclear decommissioning, and based on this, we think it will be meaningful to specify the procedures necessary for DCGL derivation.

Keywords: Decommissioning, DCGL, RESRAD, Dose assessment

Remote Monitoring Methodology for Underwater Laser Cutting in Nuclear Facility Dismantlement

Ikjune Kim*, Jonghwan Lee, Jaehyun Ha, Dongjun Hyun, and Sungmoon Joo

Korea Atomic Energy Research Institute (KAERI)

*ikjunekim@kaeri.re.kr

The dismantlement of highly radioactive nuclear facilities, particularly those with complex geometries, requires advanced underwater laser cutting techniques. To ensure safety and efficiency in these critical operations, we developed a comprehensive remote monitoring methodology for underwater laser cutting processes.

Our methodology employs a multi-sensor approach to capture and analyze data from the cutting environment. We utilized hydrophones to record acoustic emissions and pressure sensors to measure hydrodynamic changes during laser cutting operations. This diverse data collection strategy enables a more robust monitoring process.

The proposed methodology includes the implementation of a time series database for efficient storage and retrieval of sensor data. We designed a specialized database interface that facilitates real-time remote access to cutting status information, allowing operators to monitor the process from a safe distance.

Our analytical approach involves spectral analysis of the acoustic data stored in the database. This technique allows us to identify and classify the states of the laser cutting process, including success and failure.

To validate our approach, we conducted a series of controlled experiments simulating various underwater laser cutting scenarios typical in nuclear facility dismantlement. Results demonstrate that our methodology can effectively distinguish between normal cutting operations and anomalous events, providing critical real-time insights into the cutting process.

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Keywords: Nuclear facility dismantlement, Remote monitoring, Laser cutting, Data analysis

A Study on Preliminary Considerations for On-Site Dismantling Equipment Preparation and Removal Activities for Planning of Reactor Decommissioning Process

Haewoong Kim*, Kwangsoo Park, Changgyu Kim, Taeseob Lim, Donghun Pak, Woonyun Choi,
and Chungsik Lim
Doosan Enerbility
*haewoong.kim@doosan.com

The Kori Unit 1 nuclear power plant has been permanently shut down since 2017, and the dismantling of the activated reactor pressure vessel is a very important critical point in the overall decommissioning process and requires a carefully planned work packages.

In order to establish decommissioning process plans for removing heavily radioactive heavy objects within a nuclear power plant, it is necessary to analyze the technology for the entire processes, including simulations and supporting work trainings for dismantling or cutting removal engineering, and derive technical considerations in advance to suit the characteristics of the site.

When dismantling large-heavy components such as a reactor pressure vessel in nuclear power plants, both radiological and non-radiological hazards must be considered. These are parts of the safety assessment processes necessary to derive and quantify uncertainties and risks in advance when establishing a nuclear reactor decommissioning plan and to apply graded approaches to the risks.

Although many decommissioning process simulations have been developed, the installation or removal of remote cutting equipment for support work in area where workers cannot easily access with high radioactivity in the overall process requires extensive preparation and detailed radiation protection measures.

Through these derivation of factor for hazard works and establishment of pre-training plans within measures, the technical considerations for performing process simulations of the decommissioning activities and work packages of the Kori Unit 1 nuclear power plant and training workers for preparing the decommissioning equipment on-site were derived.

It is expected that this study can be useful for optimizing and designing the plans of decommissioning processes and technologies for nuclear facilities.

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Keywords: Decommissioning, Reactor, Radioactive waste, Safety, Protection, Consideration

Introduction to the Decommissioning Cost Estimate Structure of the DeCAT Program

Chan-Geun Park* and Hyung-Woo Seo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*cksrmsdl@khnp.co.kr

Kori Unit 1 and Wolsong Unit 1 nuclear power plants are being prepared for decommissioning at the end of their design life, and a Final Decommissioning Plan (FDP) must be submitted in accordance with the Nuclear Safety Act. According to the FDP's regulatory guidelines, in Chapter 2 of the decommissioning plan, project management, the cost required for decommissioning nuclear power facilities must be quantitatively evaluated in consideration of the facilities and site characteristics.

Looking at overseas cases, various computer programs are used to evaluate decommissioning cost estimate such as POWERtool, COSTGEN, PRICE, etc. In Korea, a computerized program (Decommissioning Cost Assessment Tool, DeCAT) for quantitative decommissioning cost estimate is being used. The DeCAT program is a cost evaluation program for decommissioning nuclear power plants jointly developed by Korea Hydro & Nuclear Power (KHNP), Korea Electric Power Corporation (KEPCO E&C), and Radiation Safety & Control Services (RSCS) of the United States. The DeCAT program evaluates decommissioning cost estimate based on the Work Breakdown Structure (WBS) and Unit Cost Factors (UCF) recommended by AIF/NESP-036 and the U.S. Department of Energy. According to the AIF/NESP-036 decommissioning cost estimate methodology, each decommissioning activity consists of a cost component, and the UCF determines the cost element by the unit cost of the field work.

Decommissioning cost estimate is a licensing matter for decommissioning approval and requires continuous and systematic management. Therefore, this paper introduces the work flow of DeCAT decommissioning cost estimate and shows the input data for evaluation.

Keywords: Decommissioning, DeCAT, Work breakdown structure, Unit cost factor

Underwater Laser Cutting Process Monitoring Method

Jonghwan Lee*, Dongjun Hyun, Jaehyun Ha, Ikjune Kim, and Sungmoon Joo

Korea Atomic Energy Research Institute (KAERI)

*jhl@kaeri.re.kr

Laser cutting technology has the advantage of having no direct contact with the cutting object, so there is almost no reaction force during the cutting process, and it is relatively fast, unlike other mechanical cutting technologies. Therefore, it is advantageous for tasks that are easy to work remotely, such as dismantling nuclear facilities, while improving economy and exposure safety.

In general, when working in the field, it is necessary to monitor the progress of a process to ensure that it is being carried out correctly. This is because if errors in the process are not immediately identified, unnecessary time and costs are significantly increased, and safety is compromised. However, monitoring the underwater laser cutting process poses additional constraints compared to cutting in air. The underwater environment limits the installation and operation of the equipment, the sparks from the heat source generate large amounts of bubbles from the auxiliary gases, and the longer process time results in significantly higher turbidity from the cutting byproducts that are scattered in the water, making it difficult to visually observe the work area.

This study introduces a method for real-time monitoring of underwater laser cutting processes that overcomes these limitations. By monitoring the state of the flow that occurs when a high-pressure auxiliary gas is ejected underwater, this technology can distinguish between normal and abnormal cutting conditions. This overcomes the limitations of underwater cutting monitoring, minimizing the need for complex device configurations or additional equipment, and enables the normal cutting of workpieces to be monitored during the underwater cutting process. This technology is expected to enable operators to immediately identify and respond to abnormal cutting conditions, preventing further losses due to process errors.

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Keywords: Underwater laser cutting, Process monitoring, Nuclear dismantling, Remote cutting

Analysis of Safety Assessment Methodology of Decommissioning NPP by IAEA DeSa Project

Geon Woo Son, Chang Hee Han, Seung Beom Yoo, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

In Korea, NPP Kori #1 and Wolsong #1 have been permanently shut down, and procedures for approval of the decommissioning are in progress. IAEA has stated that the decommissioning plan is most important for the safe decommissioning of nuclear facilities, and the core of the plan is safety assessment. However, there is still no systematic decommissioning safety assessment methodology in Korea. Therefore, in this study, the safety assessment methodology proposed by IAEA was analyzed to derive a systematic safety assessment methodology applicable to Korea. In order to derive a harmonious and powerful decommissioning safety assessment methodology, IAEA carried out the DeSa project.

As a result of the project, six-step methodology was derived. Each step is performed sequentially, but the results of subsequent steps can be used in previous steps as they have interdependent and repetitive characteristics. The first step is establishing a safety assessment framework. In this step, assessment system is established to ensure the reliability of decommissioning activities. These frameworks include decommissioning strategies, assessment objectives, and assessment periods. The second step is description of facility and decommissioning activities. It is a step to explain the residence information of nearby facilities and residents that may be affected during decommissioning activities. The third step is hazard identification and screening. In this step, the radioactive and non-radioactive hazards that may occur in the decommissioning activity are identified and classified according to the risk level. The fourth step is the hazard evaluation. In this step, the risk to the decision group is evaluated for the hazards selected in the previous step. The fifth step is the engineering assessment. It is the step of evaluating whether the structures, systems and components (SSCs) used in the decommissioning activity ensure sufficient safety. The sixth step is evaluation of results and identification of safety. In this step, the necessary safety control measures are applied to reduce the risk level of the previously selected hazard, and the mitigated risk level is evaluated as a result.

A graded approach must be applied throughout the entire process to efficiently utilize the resources. The results of this study are considered to be used as base data when preparing safety assessment items in the decommissioning plan for approval of the decommissioning of nuclear power plants.

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Keywords: NPP decommissioning, Safety assessment, DeSa project, Graded approach

Establishment of Infrastructure for Demonstration of Full-System Decontamination Technology

Do-Yeob Kim*, Hyeok-Jae Kim, and June-Hyun Kim

Korea Research Institute of Decommissioning (KRID)

*kdy@krid.re.kr

Full-system decontamination (FSD) is a technology to remove radioactive materials present in equipments and pipes in the primary system of nuclear power plant (NPP), and it can reduce radiation exposure to workers during decommissioning of NPP. According to the Article 85-11 (Decommissioning Strategy, etc.) of the Regulation on Technical Standards for NPP, Etc., ‘Proven technology must be applied to decommissioning of nuclear reactor facilities... (the rest omitted)’. However, in Korea, there is a lack of infrastructure such as demonstration facilities or documented procedures for demonstration to confirm that the system decontamination technology is applicable to NPP. As a result, it may be difficult to apply the FSD technology developed in Korea to the field or export it overseas in the future.

To overcome this situation, the Korea Research Institute of Decommissioning (KRID) is establishing an infrastructure for demonstrating nuclear decommissioning technology. In particular, in the case of FSD, demonstration requirements and procedures are developed, and demonstration facilities are also constructed. The demonstration requirements are divided into the design requirements and performance requirements of the equipment, and the procedure consisted of design suitability review, factory inspection, and disposal suitability evaluation. Finally, in the case of facilities, a Mock-up for light water reactor system and a number of equipment for FSD will be built in the Mock-up test building as facilities to demonstrate FSD. Mock-up, which is a model of the core and internals of NPP reactor, on a scale of 1/N, and various facilities for manufacturing chemical decontamination agents and corrosion specimens are scheduled to be built.

The demonstration procedure being developed by KRID is expected to be reviewed by a number of experts in related fields from next year to secure its validity. Furthermore, it is expected that the infrastructure of KRID will be utilized in the demonstration process of FSD during the decommissioning of NPP.

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Keywords: NPP decommissioning, Full-system decontamination, Demonstration, Infrastructure

Derivation of Demonstration Requirements for the Introduction of Abrasive Decontamination System for the Decommissioning of Nuclear Power Plants in Korea

Hyeok Jae Kim*, June Hyun Kim, and Do Yeob Kim

Korea Research Institute of Decommissioning (KRID)

*hjae5312@krid.re.kr

In Korea, starting with Kori Unit 1 and Wolsong Unit 1, the decommissioning of nuclear power plants (NPPs) is about to begin in the near future. During decommissioning, a large amount of waste is generated in short period of time, however a significant amount of these are expected to be able to be the clearance level after decontamination. In Particular, abrasive decontamination has been widely used in foreign NPPs for this purpose. For using the abrasive decontamination at the decommissioning site in Korea, it is necessary to confirm the effectiveness and validity of the technology through demonstration. In this study, as a prior research for this, the demonstration requirements of the abrasive decontamination system were developed.

In order to derive demonstration requirements, we reviewed domestic and foreign regulations and guidelines applicable to the abrasive decontamination system. According to Nuclear Safety and Security Commission (NSSC) rule, the technology applied to decommissioning of nuclear facility have to be proven to be safe and effective. Moreover, NSSC No. 2021-24 stipulates technical standards for the design and operation of radioactive waste treatment facilities. The US Occupational Safety and Health Administration (OSHA) published a guideline that presents hazards by abrasive working. In this study, we analyzed the requirements for proving the effectiveness and validity of the abrasive decontamination system, based on these.

The demonstration requirements were largely divided into 1) design requirements, and 2) performance requirements. The design requirements are applied in the design stage, and subdivided into workers/public safety, and field applicability. In the case of workers/public safety, safety requirements to be considered when designing the abrasive decontamination system were included. In the case of field applicability, the specifications and environmental conditions of the site where the system will be installed were considered. Performance requirements are mainly required by end users. we considered not only the decontamination performance such as etching rate and decontamination rate, but also the abrasive recycling rate for minimizing secondary waste, in this study.

Clearance through abrasive decontamination is related to decommissioning licensee, radioactive waste disposal licensee, and regulatory agencies. Therefore, the above requirements will be finalized after review by experts in each field. Then, these will be reflected when developing demonstration procedures. The results of this study are expected to be used as a basis for developing the demonstration system for abrasive decontamination for decommissioning waste.

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Keywords: Decommissioning, Abrasive decontamination, Demonstration, Clearance

A Study on the Evaluation of Cs-137 Adsorption Considering Soil Porosity in the Decommissioned Nuclear Power Plant Site

Do Won Hyun, Sang Heon Lee, and Jong Soon Song*

Chosun University

*jssong@chosun.ac.kr

Thorough monitoring and safe treatment and disposal are essential because soil waste generated during the site restoration phase of nuclear decommissioning has a large impact on the surrounding environment. In particular, Cs-137 can remain in the soil for a long time due to its high mobility and long half-life as a characteristic of the nuclide, and its diffusion and adsorption properties vary greatly depending on the physicochemical properties of the soil. This study was conducted in two stages with the aim of elucidating the geological characteristics of domestic nuclear power plant sites and the diffusion and adsorption mechanisms of contaminants in soil. First, the soil particle size distribution and Cs-137 adsorption characteristics in clay minerals at domestic nuclear power plant sites were investigated. The results of the study showed that the soil particle size distribution at domestic nuclear power plants showed different distributions depending on the site, and this was confirmed to be a factor that significantly affects the adsorption and distribution of Cs-137. Secondly, to evaluate the porosity of the soil and the diffusion and adsorption characteristics of Cs-137 in relation to it, a diffusion model of contaminants in the soil was constructed by reflecting the porosity and retardation factor in the advection-diffusion equation, and the results were derived to used MATLAB program. As a result, the tendency of contaminants to diffuse rapidly into deeper layers in soil with large porosity was confirmed through the advection-diffusion equation. This study analyzed the soil waste characteristics and contaminant diffusion mechanism in the site remediation phase of a nuclear power plant decommissioning site. Based on this, the diffusion characteristics of contaminants according to the physicochemical properties of soil were identified, and it is expected that this will be used as important basic data for establishing safe and efficient soil waste management and disposal strategies in the future nuclear power plant decommissioning process.

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Keywords: Site restoration, Cesium adsorption characteristics, Porosity

Analysis of SAT Based NPP Decommissioning Education and Training System

Minhee Kim* and Moo keun Song

Korea Research Institute of Decommissioning (KRID)

*kmh0408@krid.re.kr

As the demand for nuclear power plant (NPP) decommissioning at domestic and overseas steadily increases, the need for systematic worker education and training is increasing. The US 10 CFR 50.120 provides guidance on education, training, qualifications for nuclear workers. ANSI/ANS-3.1 produces standards for training and qualification requirements for workers. Accordingly, DOE-HDBK-1078 published the systematic approach to training (SAT) guidelines to provide guidance for establishing education and training. SAT is also being applied domestically to education and training in the construction and operation of NPP. However, in the field of NPP decommissioning, most of the training is theoretical, and SAT is not reflected in training that takes into account the actual work environment.

Accordingly, this study aims to establish a systematic education system in the field of NPP decommissioning that takes into account the 5 stages of SAT. The SAT 5 stages consist of analysis, design, development, implementation, and evaluation. The analysis stage is divided into necessity analysis, job analysis, and work analysis. In this study, a demand survey was conducted targeting approximately 120 people in charge of NPP decommissioning companies and institutions. The standard error of this demand survey is the maximum allowable sampling error of 9.2%p at a 95% confidence level. According to the survey results, 88.5% of respondents said they needed 'Theory and practical training' for NPP decommissioning. The highest goal for education using virtual reality/augmented reality (VR/AR) was 'Acquiring information and knowledge about NPP decommissioning scenarios' at 49.6%. Next, 'Understanding the internal structure of a NPP' was at 23.9%, 'Dismantling preparation and post-processing work' was at 23%, and 'How to use cutting equipment for NPP dismantling' was at 3.5%. In addition, expert opinions were reviewed on the level-specific training hours and content for 'Decommissioning process training' and the role-specific composition and training level for 'NPP decommissioning field work support training'. Finally, evaluation factors were derived to assess trainees after completing the training.

Based on this, detailed job analysis will be conducted, and based on the results, data on the knowledge, skills, and attitudes required for each element of NPP decommissioning training will be secured. After that, an evaluation of each work scenario will be conducted. This is expected to become a foundation for the growth of NPP decommissioning education in the future.

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Keywords: NPP, Decommissioning, SAT, Education, Training, VR, AR

Development of a Mobile Decontamination System for Both Modular and On-Site Installation for Radioactive Liquid Waste Treatment at Nuclear Facilities

Jihyun Kang^{1,*}, Choongsup Byun¹, Ilgook Kim², In-Ho Yoon², Suin Lee¹, Wonseok Park¹, Dohyung Kim¹,
and Jongyoul Lee¹

¹Beautiful Environ. Construction Co., Ltd.

²Korea Atomic Energy Research Institute (KAERI)

*jhkang@esakorea.com

Liquid waste generated during the operation of a nuclear power plant can be treated through the radioactive liquid waste treatment system in the plant, but in the event of an accident such as the Fukushima Daiichi Nuclear Power Plant accident or decommissioning, it may be difficult to treat it in a timely manner with a fixed radioactive liquid waste treatment facility. Therefore, we aimed to develop a mobile radioactive liquid waste treatment system that can respond immediately on site.

The developed mobile radioactive liquid waste treatment system consists of a pretreatment stage and a main treatment stage. The pretreatment stage consists of a Hollow Fiber Filter (HFF) and a Magnetic Separator (MS) to filter out large particle size suspended solids and fine soil, as the performance of the downstream decontamination unit is degraded when water is collected from outdoor lakes. The main treatment stage consists of Reverse Osmosis (RO) and Ion Exchanger (IX). Each stage is modularly installed in a separate container so that each process can be operated independently. As a key feature of the system, MS incorporates magnetic nano-particle adsorbent to adsorb colloidal substances that are less filtered out by HFF and radionuclides, which can reduce the load of RO and IX with moderate decontamination performance and minimize the generation of secondary waste. The system is also designed to be operated in a modular fashion, with a combination of decontamination unit selected according to the radioactivity concentration of the incoming water to minimize the frequency of ion exchange IX resin replacement, and the final decontaminated water is only transferred to the final treated water tank if it meets the discharge criteria via the effluent radiation monitor.

The performance targets of the system were to have a capacity of more than 0.5 m³/h with a decontamination rate of 90 to 99% for the representative radionuclides Cs, Sr and Co and an effluent turbidity below the drinking water quality standard of 1 NTU. The developed facility was manufactured to have a capacity of 1 m³/h, and the performance tests confirmed a decontamination rate of more than 99% for all nuclides, and the turbidity of 0.24 NTU, which sufficiently exceeded the target.

In the future, it will be possible to maximize the performance of this system by developing magnetic nano-particle adsorbents that can perform high decontamination performance for each nuclide and conducting experiments to derive optimal operating conditions.

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Keywords: Liquid radwaste, Decontamination, Mobile LRS, Magnetic separator

Review on the Parameter Selection Process in the Probabilistic Analysis Using RESRAD on the Site of Decommissioning Nuclear Facilities

Hyung-Woo Seo*, Young-Il Na, Chan-Geun Park, and Gang-Woo Ryu

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sehywo1727@khnp.co.kr

The decommissioning of domestic nuclear power plants (NPPs) is currently in the preparatory stage, and a Final Decommissioning Plan (FDP) for approval of decommissioning is being reviewed. The ultimate goal of decommissioning NPPs is to release the site and the remaining buildings. Dose criteria is in place for deregulation of facilities, but the corresponding residual radioactivity concentration so called Derived Concentration Guideline Level (DCGL) must be calculated and presented by a licensee. Usually, the RESRAD code has been used for this purpose, and it is necessary to input parameter values in connection with the site characteristic data. However, if the designation of input parameter values become less available or need to be given as a function of the distribution, a probabilistic analysis is performed. Therefore, the purpose of this study is to review the parameter selection process they adopted in conducting probabilistic evaluation through overseas cases.

Keywords: Decommissioning, DCGL, RESRAD, Dose assessment

Review of Exposure Pathways for Evaluating Dose of the Site, Buildings, and Nearby Locations When Using the RESRAD Codes

Hyung-Woo Seo*, Young-Il Na, Chan-Geun Park, and Gang-Woo Ryu

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sehywo1727@khnp.co.kr

The final stage of the nuclear power plant (NPP) decommissioning project is the remediation of the site and should be below the release criteria for the remaining sites and buildings. Typically, the RESRAD code is used to derive Derived Concentration Guideline Level (DCGL) for sites and remaining buildings. The RESRAD codes are developed by the U.S. Argonne National Laboratory (ANL) and has been widely used to release the site of the decommissioning nuclear facilities. In addition, RESRAD-ONSITE has been used for soil media and RESRAD-BUILD has been used for remaining buildings. For sites, the RESRAD-OFFSITE code is also provided by ANL, especially for the nearby residents of contaminated sites. The concept of dose evaluation for the receptor in each code will be different, and the starting point of the contamination source will be also different. Therefore, the purpose of this study is to review the exposure pathways considered and assumed in these codes.

Keywords: Decommissioning, DCGL, RESRD-onsite, RESRD-offsite, RESRAD-build

A Case Study on Measurement Methodology in Planning Phase of Scoping Survey and Characterization Survey

Hyungi Byun¹, Jaehoon Byun¹, Juyub Kim^{1,*}, and Seokyoung Ahn²

¹FNC Technology Co., Ltd.

²Pusan National University (PNU)

*yubjoo@fnctech.com

Site characterization for the decommissioning of Nuclear Power Plants (NPPs) and related facilities is generally carried out by referring to NUREG-1575, developed in the United States. In the U.S., the results of a scoping survey and characterization survey are submitted to the U.S. Nuclear Regulatory Commission (U.S. NRC) during decommissioning. However, details regarding the selection and number of measurement points are included in the License Termination Plan (LTP), which is submitted after the planning of the Final Status Survey (FSS). In contrast, the Final Decommissioning Plan (FDP), which describes the radiological characterization of the site and the associated results, must be submitted to the regulatory body for the decommissioning approval in Korea.

Therefore, it is necessary to develop a validation methodology from a regulatory perspective during the scoping and characterization surveys. This study aims to provide evidence for establishing a procedure to decide the measurement (i.e. instrument, method, point and quantity) based on Data Quality Objectives (DQO). We reviewed overseas cases on the decision making process regarding measurement according to the scoping and characterization surveys during the decommissioning process.

In case of Maine Yankee NPP, where the MARSSIM methodology was applied for the first time, the survey area was divided into 7 groups (Group A, B, C, D, R, E, and H), with measurement instruments chosen based on specific Radionuclide of Concern (ROC) and the residual radioactivity to be measured. And the scoping survey of Zion NPP was planned to determine measurement points and their quantities to cover the expected contaminated areas. This planning was based on the results of the Historical Site Assessment (HSA), which was previously performed to set the survey extent. On the other hand, a characterization survey plan of Rancho Seco NPP was established to further divide the contamination areas identified in the scoping survey, determine their radiological characteristics, and measure residual radioactivity. In addition, cases from various countries were analyzed.

Consequently, the procedure for decision making regarding measurement may vary depending on the purpose of the survey plan, which should be established based on DQO and guarantee the safety from a regulatory perspective. Therefore, the measurement method should be planned according to the radiological characteristics and residual radioactivity. And when statistical test methodologies (e.g., WRS test and Sign test) were applied, it will be possible to serve as the basis for determining the selection of measurement points and their quantities.

Acknowledgements

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Keywords: Nuclear decommissioning, Radiological characterization, Survey point, Data quality objectives, Scoping survey, Characterization survey

Effect of Jahn-Teller Distortion on Iodine Sequestration of CuAl Layered Double Hydroxides

Jueun Kim and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongum@postech.ac.kr

The Jahn-Teller effect is a geometric distortion of a non-linear molecular system, which reduces its symmetry and energy. This effect is dependent on the electronic state of the system and occurs when ions have an asymmetric electron count. The d^9 electron configuration of Cu^{2+} ions gives rise to the formation of a distorted octahedral structure as a consequence of the Jahn-Teller effect. These properties render CuAl layered double hydroxides (LDH) more structurally unstable than other LDHs. The objective of this study is to optimize the synthesis method of CuAl LDH and to investigate the feasibility of iodine sequestration while simultaneously stabilizing the CuAl LDH, which is inherently unstable.

CuAl LDH was synthesized at varying pH levels (8~11) and aging temperatures (20~80°C). The formation of the copper hydroxyl complex, rouaite, was observed at low pH and aging temperatures, while CuO was formed at high pH and aging temperatures. The optimal synthesis condition was found to be pH 11 at the aging temperature of 40°C. Iodide sorption experiments were performed with 10 mM NaI solution at a solid to solution ratio of 10 g/L. CuAl LDH sample was observed to remove 724.9 mg/L of iodide, which is equivalent to 5.71 mmol/L of I^- . No significant changes were observed in the XRD pattern before and after iodine adsorption. However, following a heat treatment at 120°C for one day, CuAl LDH containing iodide underwent a transformation, resulting in the formation of CuO with the partial conversion to CuI. Despite heating the CuAl LDH containing nitrate to 120°C, the XRD pattern remained similar, and no observable peaks corresponding to CuO were observed. Due to the Jahn-Teller effect, CuAl LDH is metastable and decomposes at a lower temperature compared to other LDHs, although CuAl LDH containing nitrate does not decompose at 120°C. However, partial of Cu^{2+} in the CuAl LDH containing iodide was reduced to Cu^+ and resulted in the formation of CuI through the equation, $2\text{Cu}^{2+}(\text{aq}) + 4\text{I}^- \rightarrow 2\text{CuI}(\text{s}) + \text{I}_2(\text{aq})$, in which iodide played a role in facilitating the reduction of Cu^{2+} to produce CuI.

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Keywords: Layered double hydroxides, Jahn-teller effect, Iodine, Sequestration

Case Study of Characteristics of Radioactive Aerosol Generation Based on Cutting Media and Filtration System

Donghun Pak*, Kwangsoo Park, Chungsik Lim, Changgyu Kim, Haewoong Kim, Woonyun Choi,

and Taeseob Lim

Doosan Enerbility

*donghun.pak@doosan.com

The decommissioning of nuclear power plants is an essential process that must be carried out in the near future by countries utilizing nuclear power. Among the various tasks required during decommissioning, it is crucial to safely cut or separate and manage metal materials that have become activated due to the prolonged operation of the nuclear power plant. During the cutting of these activated metals, various secondary by-products are generated simultaneously, among which aerosols, composed of very fine particles, are identified as a major cause of internal exposure to workers. Therefore, it is imperative to understand the characteristics of these aerosols thoroughly and to prepare appropriate radiation protection measures as a prerequisite for the decommissioning process.

This study analyzes the radioactivity characteristics of aerosols generated during thermal cutting processes and proposes fundamental protective measures to minimize worker exposure. Aerosols produced during the cutting of activated metals vary in size, and their distribution in size and concentration is influenced by factors such as cutting method, type of metal, and cutting conditions. These aerosols' physical and chemical properties significantly affect the radiation dose to which workers are exposed, potentially increasing the risk of internal exposure. This study provides a comprehensive analysis of these aerosol characteristics, offering foundational data for establishing effective radiation protection strategies in the workplace, while also discussing optimization methods for protective equipment such as filtration systems.

In conclusion, radiochemical understanding of aerosols generated during the nuclear dismantling process and subsequent radiation protection measures are essential to protect workers' health and ensure the dismantling process's safety.

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Keywords: Aerosol, Decommissioning, Radionuclide, Filtration system

Regeneration of Spent Ion Exchange Resin Using a Bipolar Membrane Cell

Sang Yoon Park^{1,*}, Byoung Seon Choi^{1,2}, Shin Yoeng Lee¹, Byung Gwan Kang¹, and Won Keun Son¹

¹Innochemtech

²Korea Atomic Energy Research Institute (KAERI)

*nsypark@innochemtech.com

Ion exchange resins (IERS) are used to control the chemistry of the system water, minimize corrosion and remove radioactive contaminants during the commercial operations of NPPs as well as nuclear decommissioning. Significant amounts of spent ion-exchange resins (SIERs) have stored in the nuclear power plants (NPPs) due to power operation and still being produced from NPPs. The reduction of SIERs and its treatment to the forms suitable for long-term disposal is an urgent problem facing the nuclear power generation. SIERs are not only difficult to treat due to their large amount and high water contents but also inefficient for long-term storage due to their large volume. These spent resins should be properly treated and disposed in order to minimize their potential hazard to the environments. Thus, the regeneration of SIERs containing various radioactive nuclides, such as Cs-137, Sr-90, Co-60, Ru-106 and Mn-54, would therefore produce large volumes of liquid waste but this step is so complicated that SIERs are typically replaced without any reuse. The resins loaded with radioactive nuclides could not be regenerated and reused.

The final goal of this study is to regenerate and reuse spent ion-exchange resins by the most environmentally safe way without any release radioactive wastes. As a first step, we tested a method to generate SIERs using a bipolar membrane (BPM) cell that does not generate liquid waste at all. In particular, the regeneration of SIERs was evaluated by considering the adsorption and desorption characteristics of the Fe(III)-EDTA complex by cation and anion exchange resins.

Keywords: Spent Ion Exchange Resins (SIERs), Decommissioning, Bipolar Membrane (BPM), Adsorption, Desorption, Nuclear Power Plants (NPPs)

Development of a Mobile Scattering Dust Removal System Using Plasma Technology to Manage Dust Generated From NPP Decommissioning Sites

Dong-Gyun Lee and SeokYoung Ahn*

Pusan National University (PNU)

*sahn@pusan.ac.kr

In Korea, some decommissioned nuclear power plants are preparing for dismantling. Considerable research efforts are underway to ensure that the dismantling process is conducted both safely and efficiently. Moreover, the cost associated with the disposal of radioactive waste is approximately 15.11 million Korean Won per 200 L drum, imposing a significant financial burden on the decommissioning operators. Therefore, decontamination of contaminated facilities should be performed to reduce waste disposal costs during decommissioning. This process may result in the generation of various types of radioactive dust. Particularly, workers operating in confined spaces, such as the interior of containment vessels where air purification is challenging, may experience increased radiation exposure, leading to potential threats to the surrounding community and environment. To solve this problem, a mobile scattering dust removal system capable of capturing and processing dust generated on-site according to the decommissioning procedure developed. This system employs plasma pyrolysis technology for the active treatment of gaseous effluents and is designed as a modular, movable unit for ease of transportation and reuse. Through this system, the removal of hard-to-degrade toxic substances, including radioactive dust, is achievable, thereby fundamentally preventing the occurrence of secondary contamination sources. This approach is anticipated to significantly reduce decommissioning costs.

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Keywords: Decommissioning, Plasma, Mobile, Dust

A Study on Safe Dismantling Methods and Sequences for Asbestos-Containing Insulation in the Turbine Building of Kori Unit 1

ChungSik Lim*, Kwangsoo Park, Changkyu Kim, HaeWoong Kim, Woonyun Choi, Donghun Pak,

Taeseob Lim, and Nari Kim

Doosan Enerbility

*Chungsik.lim@doosan.com

Kori Unit 1 nuclear power plant is a pressurized water reactor type with an output of 587 Mwe, which was permanently shut down on June 18, 2017. Currently, the final decommissioning plan (FDP) has been submitted and review is in progress.

We are considering the dismantling of major equipment in the turbine building of Kori Unit 1 before the approval of the Final Decommissioning Plan (FDP), particularly focusing on the removal of insulation material that wraps around the piping for steam insulation.

It is estimated that asbestos, which is classified as a carcinogen, has been used as insulation material in Kori Unit 1. The dismantling of insulation material containing asbestos must comply with the Asbestos Safety Management Act for waste management and disposal.

Asbestos continued to be used in industrial sites due to its easy procurement and low cost. However, its use has been gradually restricted since 2007, with the full implementation of the Asbestos Safety Management Act on January 1, 2009, which prohibits the manufacture, import, and use of products containing more than 0.1% asbestos.

Since Kori Unit 1 has been in commercial operation since 1978, it is expected that some asbestos-containing insulation materials were used during construction. Prior to dismantling the turbine building, it is essential to conduct a thorough investigation and organize the related data to establish and execute a safe dismantling plan for the insulation materials wrapping the steam pipes.

Piping that may contain asbestos insulation covers a wide range from the outlet to the inlet line of the primary system steam generator, with a large quantity installed. The dismantling plan must address various environments, including high and low positions, large and small pipe sizes, and favorable and unfavorable insulation removal positions.

If the insulation materials installed in these diverse environments are managed meticulously using appropriate dismantling sequences and methods, the likelihood of workers inhaling asbestos during the process can be minimized, and external contamination can be prevented.

The purpose of this study is to summarize the research on safe dismantling methods and sequences for asbestos-containing insulation material in the turbine building of Kori Unit 1.

Keywords: Dismantling, Asbestos, Insulation material, Safety management, Dismantling plan

Radiation Monitoring Based Radioactive Contaminated Soil Decontamination System

Seongjoo Kang*, Myeonghoon Joo, Sangwook Lee, Youngjoo Song, Doocheol Kim, Sanghyun Jeong,
and Jiyang So
DSENTEC Inc.

*seongjoo.kang@dsentec.com

We developed a high-pressure washing-based mobile radioactive contaminated soil decontamination system to treat soil contaminated by radioactive materials at the nuclear dismantling site or at the nuclear site. In evaluating the decontamination efficiency of the present technology, there is a need for a system capable of quickly measuring the radiation and radioactivity of radioactive contaminated soil.

In this study, a monitoring system for field measurement was fabricated using a plastic scintillator with a uniform and sensitive reactivity to gamma rays to reinforce the radiation measurement system at the outlet of the drying system, which is the rear end of the process. Radiation linearity and Compton edge of the gamma spectrum were confirmed using standard radioactive sources (Cs-137, Co-60).

In addition, performance tests were conducted through comparative analysis with portable NaI detectors and HPGe gamma nuclide analyzers in our RI laboratory to confirm the possibility of field application.

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Keywords: Radioactive soil, Radiation monitoring, Soil decontamination

Development of Equipment for Sampling From Radioactive Waste Drums

Hye Jin Kim, Jung-Min Oh, Wook Jae Yoo, Kyu-Tae Park, Bo-Bae Cho, Sung-Ryul Kim, Ji-Ung Kim,
Ba-Ro Lee, Hyun-Young Shin, So-Yun Jeong, and Chan-Hee Park*

Orbitech Co., Ltd.

*chpark@orbitech.co.kr

Currently, the post-closure safety assessment in the radioactive waste disposal facility relies on international data under conservative assumptions due to the uncertainty regarding the various types of waste and the stored materials. This approach can lead to overly conservative dose assessment results, which may reduce the amount of radioactive waste that can be disposed of and negatively impact the efficient operation of disposal facilities. The ultimate aim of this study is to obtain basic data for more accurate safety assessments by categorizing and grouping radioactive waste according to its source, type, and form, and by analyzing its physical properties. In order to experiment for the characteristic analysis such as physical properties (density, porosity, and etc.), leaching, diffusion, and hydraulic conductivity, it is necessary to fabricate the simulated radioactive waste drum generated using the 200 L drum and the samples extracted from it. To achieve this, we have developed device that can obtain sampling waste from the simulated 200 L drum.

Unlike existing core sampling equipment for solidified waste, the developed device is designed for sampling solid waste of various materials, such as paper, plastic, and rubber. It is capable of coring the lid of a 200 L drum in a single operation without the need for additional equipment changes. Furthermore, the device includes a manual lift that allows it to move vertically and horizontally, enabling precise sampling from the desired location within the drum. Additionally, a specially designed accessory called the 'hole cutter guide' can be attached to the drum lid to minimize vibration during coring, ensure consistent coring, and improve the representativeness of the collected samples by allowing sampling from the same location each time.

By using this equipment to directly core and sample from 200 L drums, it is possible to create experimental mock-up samples of various materials. The results of the characteristic analysis experiment derived from these samples can be used as foundational data to enhance the reliability of radioactive waste management and safety assessments, thereby contributing to the efficient operation of disposal facilities.

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Keywords: Radioactive waste, Waste disposal facilities, Safety assessment, Coring equipment, Drum sampling

Optimization of Decontamination Equipment for the Treatment of Radioactive Waste Generated During Decommissioning of Nuclear Power Plant

Suk Bon Yoon*, Hyun Min Kim, Sukwon Jung, Gangwoo Ryu, and Jin-Won Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sbyooncri137@khnp.co.kr

When decommissioning a nuclear power plant, proven technologies must be applied in accordance with relevant laws and regulations, and in particular, the introduction of decontamination equipment with high decontamination effectiveness is necessary to maximize the clearance of radioactive waste of large quantities of metallic radioactive waste.

Currently, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) is in the process of a constructing related treatment facility to process the large amount of radioactive waste generated during the decommissioning of nuclear power plant. The treatment facility is expected to be equipped with various facilities such as cutting, decontamination, reduction, and measuring equipment.

Among the various types of radioactive waste generated in large quantities during the decommissioning of nuclear power plants, metal waste is generated including radioactive waste such as reactor vessel and reactor vessel internal, as well as and medium-sized metal waste such as metal cuttings, valves, and fittings. Accordingly, in order to increase the decontamination effect on metal waste, it is necessary to optimize the equipment applied to each decontamination target.

To this end, this study reviewed cases of construction and operation at overseas decommissioned radioactive waste treatment facilities and, based on the review results, sought to derive an optimization plan for related equipment for decontamination of domestic decommissioned radioactive waste.

Keywords: Decommissioning, Optimization, Decontamination Equipment, Radioactive waste

Review of Domestic and International Operating Cases for the Establishment of a Decommissioned Radioactive Waste Treatment Facility

Suk Bon Yoon*, Hyun Min Kim, Sukwon Jung, Gangwoo Ryu, and Jin-Won Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sbyooncri137@khnp.co.kr

It is expected that about 80,000 drums of various types of radioactive waste will be generated during the decommissioning of a nuclear power plant. Accordingly, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) is applying various treatment technologies to reduce the large amount of radioactive waste generated during the decommissioning of a nuclear power plant to less than 14,500 drums.

To achieve these goals, it is absolutely necessary to reduce the volume of solid waste, which accounts for the largest amount of decommissioning waste. Currently, KHNP is promoting the construction of related facilities to treat large quantities of radioactive waste generated during the decommissioning of a nuclear power plant. The treatment facility will be equipped with various facilities such as cutting, decontamination, reduction, and measuring equipment, and in particular, it is planned to build a metal melting facility, which is one of the facilities for reducing solid waste.

Accordingly, as a result of conducting a domestic and international case study on the relevant facilities, it was found that AIM and VIM were in operation. Studsvik, Siempelkamp, and EnergySolutions were found to be using metal melting facilities to treat and recycle metal waste. Related facility at KNF are in operation for decontamination and volume reduction of radioactive waste, and KAERI is planning to install and operate related facility to treat metal waste at the integrated radioactive waste management facility.

Since the AIM has numerous licensing and operating cases both domestically and internationally, it is believed that it will be possible to secure various information necessary for the construction and operation of the relevant facility in the future. However, it is believed that care must be taken in the safety management of workers, such as radiation exposure management and industrial safety, when operating the relevant facility. On the other hand, although the VIM may be good for worker safety management during facility operation, it requires a lot of maintenance due to failures caused by remote facility operation, and there is a possibility of generating environmentally hazardous substances such as NO_x because it uses inert gases such as nitrogen.

Based on the results of these domestic and international case studies, we are planning to build AIM within a decommissioned radioactive waste treatment facility, and are currently pursuing detailed design for the facility.

Keywords: Decommissioning, Radioactive waste treatment facility, Induction melter

Microwave Treatment System for Activated Carbon Radioactive Waste for Upcycling Technology of B₄C Neutron Absorber

Ki Rak Lee*, Hwan-Seo Park, and Ga Yeong Kim

Korea Atomic Energy Research Institute (KAERI)

*kirakki@kaeri.re.kr

In NPP (nuclear power plant), boric acid is used as a neutron absorbent. So radioactive boric acid waste are generated from various waste streams such as discharge or leakage of reactor coolant water, floor drains, drainage of equipment for operation or maintenance, reactor letdown flows and etc. Depending on KHNP, 20,015 drum (200 L drum) of concentrated boric acid waste were stored in KOREA NPP until 2019. In previous study, The Korea Atomic Energy Research Institute (KAERI) has developed a conceptual technology to upcycle boric acid-containing radioactive waste into neutron absorbers (B₄C containing materials) for criticality control, which is necessary for the dry storage of spent nuclear fuel. The technology concept comprises a pretreatment process for radioactive waste-activated carbon to supply C raw material, a manufacturing process for B₄C using radioactive waste, and a process for producing B₄C supports using metallic radioactive waste. In order to utilize waste-activated carbon as a raw material for manufacturing neutron absorbers for criticality control, it is essential to develop a pretreatment process that removes radionuclides such as C-14 and H-3 contained within the waste-activated carbon. KAERI had developed a kilogram-scale pilot system for radioactive waste-activated carbon and had successfully treated around 30 kg of actual waste-activated carbon through demonstration. Additionally, the potential for recycling the treated waste-activated carbon was evaluated, and methods for recycling were proposed.

Separation of Hydrogen Isotopes Using Platinum Incorporated Mesoporous Silica Composite With Hydrophobic Polymer via the Liquid Phase Catalytic Exchange System

Daeuk Kang^{1,2}, Chan Woo Park², and Hyung-Ju Kim^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*hyungjukim@kaeri.re.kr

Tritium is a radioactive isotope of hydrogen with a half-life of 12.3 years, and emits beta ray. In South Korea, tritium is produced in heavy-water reactors, such as the Wolsong Nuclear Power Plant, where heavy water (D₂O) is used as a moderator and a coolant. The deuterium in D₂O can be converted into tritiated water (DTO or T₂O). To handle tritiated water, the Wolsong Tritium Removal Facility (WTRF) plays a crucial role. WTRF carry out combination techniques of liquid phase catalytic exchange (LPCE) and cryogenic distillation for the separation of hydrogen isotopes. However, since tritiated water has remarkably similar physicochemical properties as natural water, the LPCE process encounters significant challenges. Furthermore, the Fukushima nuclear accident brings worldwide attention to the importance of tritium removal, highlighting its ongoing needs for research in this field.

In this study, we synthesized a catalyst by incorporating platinum into mesoporous silica by hydrogen reduction. The mesoporous silica possesses a high surface area and a 3D interconnected porous structure favorable to the catalytic reactions. The prepared catalyst was subsequently mixed with polydimethylsiloxane to form composite and impart hydrophobicity and ensure its catalytic stability during operation. The catalyst composite was evaluated in the lab made LPCE apparatus to separate trace amount of deuterium (147 $\mu\text{mol mol}^{-1}$) from deionized (DI) water. For this purpose, the column was randomly packed with the catalyst composites and Dixon rings (hydrophilic filler) in a specific volume ratio. During the catalytic exchange reaction, DI water was slowly dripped from the top of the column while pure hydrogen gas (99.999 %) was introduced from the bottom of the column.

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Keywords: Hydrogen isotope, Tritium, Separation, Liquid phase catalytic exchange, Mesoporous silica

Radiological Impact of Radioactive Waste Treatment Facility Accidents During Nuclear Power Plant Decommissioning

Hyeonyeong Lee* and Chang-Lak Kim

Kepco International Nuclear Graduate School (KINGS)

*kkzz2580@naver.com

While many studies have been conducted on public dose assessment for accidents during nuclear power plant operation, research on worker exposure in hypothetical accidents during the decommissioning phase of multi-unit sites has been limited. This study aims to evaluate the radiation exposure dose for external workers at adjacent operating units in case of an accident at a radioactive waste treatment facility located within a site undergoing decommissioning. For the assessment, four years (2020-2023) of meteorological data from the Kori Nuclear Power Plant (Kori NPP) site were utilized. The PAVAN program was used to calculate atmospheric dispersion factors for the initial two hours after a hypothetical accident, and the radiological impact assessment for adjacent units was based on the maximum atmospheric dispersion factor of gaseous radioactive materials released during the accident. Radioactive nuclides ^{60}Co , ^{137}Cs , ^{90}Sr , and ^{65}Zn were selected based on the results of a dose rate assessment conducted for the steam generator replacement of Kori unit 1 in 1998. The exposure doses of the selected radioactive nuclides were evaluated at various points near the waste treatment facility. Evaluation points included Kori units 2, 3, 4, Shin-Kori units 1, 2, and the main entrance of the Kori NPP. Results indicate that even if external workers (maintenance personnel, security staff, etc.) are present at nearby operating nuclear power plants during an accident, their radiation exposure dose is expected to remain below 1 mSv, which is the annual radiation dose limit set for the public.

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Keywords: PAVAN model, Melting furnace explosion accident, Atmospheric dispersion factor, Multi-unit nuclear power plant

Analysis of System Decontamination Effectiveness in Different System Decontamination Scope

Jeseok Song and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*thdwptjr1@naver.com

This study compares how the change in decontamination scope affects the total decontamination factor (DF) in system decontamination of nuclear power plant, by developing and using decontamination factor evaluation program (DFEP). For the study, the factors related to the DF are found, and the effectiveness of decontamination analyzed by using DFEP with those factors, in case of the effectiveness of Kori-1 decontamination, as a representative example. The DFEP calculate the total DF from initial activity inventory and its measurement date of each part in whole range of decontamination, decontamination date and number of decontamination cycles, and DF of each cycle in each part. For every cycle, DF of each part are driven by DF of each cycle in whole system, and DF of each part in whole cycles, both from overseas full system decontamination cases. In the analysis using DFEP, the exclusion of the part with high initial activity inventory and high DF largely affect the total DF. For the steam generators in Kori-1, in the case of excluding the SG connected to pressurizer side, which have less initial activity inventory, the total DF decreased less.

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Keywords: System decontamination, System decontamination scope, Decontamination factor, Steam generator

The Evaluation of CRUD Inventory in Equipment and Pipes of Decontamination Flow Path Before and After Full System Decontamination (FSD) Using the MicroShield Software Program

Hyo-kyoung Lee* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*hyo121864@gmail.com

Kori Unit 1, Korea's first PWR, is undergoing decommissioning after 40 years of operation. Starting on May 7, 2024, the country's first full system decontamination (FSD) using a chemical decontamination method is in progress over a planned period of about four months. The decontamination, which is the most preemptive process in nuclear power plant decommissioning, is an important process that not only protects workers from excessive radiation exposure during the subsequent decommissioning process, but also reduces the amount of radioactive waste generated, alleviating the burden on waste treatment facilities. Korea Hydro & Nuclear Power (KHNP) is developing and applying its own CRI_RWDecom (chemical reagent injection and rad waste decomposition) decontamination process and CIDF (chemical injection and decomposition facility) decontamination equipment.

Accordingly, the efficiency of the chemical decontamination process of Korea's first Kori Unit 1 using CRI_RWDecom technology was evaluated. Before the FSD, an initial assessment of contamination levels within the main equipment and pipes of the decontamination flow path was performed. First, when the reactor coolant of Kori Unit 1 was completely drained, the contact dose rate was measured for each of the main equipment and piping, such as RCS, CVCS, RHRS, and SG U-Tube. And a representative measurement point (RCP Suction Leg A/B), which is easy to measure on-site, was selected and CRUD radioactivity was measured on-site using a portable HPGe detector (ISOCS) at this point. It is assumed that the on-site measured radioactivity inventory per effective measurement volume of ISOCS equipment is distributed identically to the radioactivity inventory of each equipment and pipe, and the specifications and surface dose rate of each device and pipe are corrected using the MicroShield computer program. In addition, the decontamination process was comprehensively analyzed by applying a decontamination factor (DF) for final evaluation of the residual radioactivity after FSD. This study can be used to predict the effectiveness and efficiency of chemical decontamination processes before FSD is completed and to develop a comprehensive radiation protection plan for decommissioning workers.

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Keywords: Kori unit 1, Full System Decontamination (FSD), CRI_RWDecom, CIDF, MicroShield computer program, Decontamination Factor (DF)

A Study on Radiation Shielding Plan of Estimated High Radioactivity Area During Chemical Decontamination in Kori-1

Taewoong Lee*, Seongmin Jo, Kihyun Cho, Kichul Kim, Junghyun Lee, and Sunghyun Lee

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*Superdapo@kps.co.kr

The Kori-1 nuclear power plant has been permanently shut down since 2017, and the chemical decontamination is currently planned for the primary circuits until 2024. In the chemical decontamination of the primary circuits including a reactor coolant system (RCS), a radiation protection plan should be established to minimize the radiation exposure doses of decommissioning workers in estimated high radioactivity area such as ion exchange resin tank and spent filters. The radiation protection plan, according to the optimized geometric structure, transportation of radioactive waste and the limitation of dose rate in Final Safety Analysis report (FSAR) was determined. The most significant γ -ray-emitting radionuclide is Co-60 (1,173 and 1,332 keV) in the radioactive corrosion products (CRUD). In this study, the radiation protection plan for estimated high radioactivity area was analyzed using the MCNP 6.2 version. The limitation of dose rate for radiation area was determined based on the FSAR. Finally, to estimate the best performing realistic shielding material, several types of materials were tested and their shielding performance compared. The estimations of the simulation results for all radiation protection plans satisfied the dose rate limits prescribed by the FSAR. Considering working area and costs of the shielding materials, it is demonstrated that it would be more efficient to use a shielding material composed of a water shielding rather than a single shielding material. This study will be offered guidelines for the radiation safety of the decommissioning workers in chemical decontamination.

Keywords: Chemical decontamination, Kori-1 NPP, Radiation shielding, High radioactivity area, FSAR, MCNP6.2

A Study on Optimization of Monitoring System for Radiation Dose Rate During Chemical Decontamination in Kori-1

Taewoong Lee*, Mikyung Hwang, Seongmin Jo, Kihyun Cho, Kichul Kim, Junghyun Lee, and Sunghyun Lee

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*Superdapo@kps.co.kr

The performance of the process for the chemical decontamination could be estimated by measuring radiation dose rate. For the accurate verification of the performance of the chemical decontamination cycles, it is important to monitor the radiation dose rate of IX (Ion Exchanger) tank in real time during the process of the chemical decontamination. Because of the ion exchange of Co-60 radioactive isotope primarily occurs at the bottom of the IX tank, the monitoring system is located at one-third point of the height of the IX tank. During the process for the chemical decontamination, the temperature of the water in a reactor coolant system (RCS) rises above 100 degrees and the γ -rays of Co-60 are emitted from all directions in IX tank. Hence it is necessary to optimize the geometrical parameters based on the characteristic of monitoring system. In this study, in order to maximize the performance of the monitoring system, the geometrical parameters such as the distance between IX tank and detectors and hole size and septal thickness of collimator were optimized by using simulation codes with the MCNP6.2 and ANSYS 2022. According to our simulation results, for high detection efficiency and small background counts, the optimum septal thickness was set for longer than 5 cm and the distance between IX tank and detector was set to 10 cm. These results and methods demonstrated the optimized geometrical parameters for use as monitoring the radiation dose rate in the process of the chemical decontamination.

Keywords: Chemical decontamination, Radiation dose rate of IX tank, Optimized geometrical parameters of monitoring system, MCNP6.2, ANSYS2022

A Study on Estimation of Collective Dose for Pre-Treatment Workers in Separation Process of a Spent HEPA Filters Using PIMAL Phantom

Taewoong Lee*, Yongho Cho, Seongmin Jo, Kihyun Cho, Kichul Kim, Junghyun Lee, and Sunghyun Lee

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*Superdapo@kps.co.kr

Based on the real characteristics of the separation process for a spent HEPA filters, the radiation exposure dose was estimated for pre-treatment workers at different working posture using the phantom with moving arms and legs (PIMAL) in this study. The PIMAL phantom consisted of 30 anatomical regions and each cell had the same material information as a real human cell. The collective dose, which is defined as the sum of the individual effective dose to each tissue considering the working time, was used to estimate the radiation exposure dose. Moreover, during separation process of a spent HEPA filter, the radiation exposure dose for arms and legs which were close to the radiation source was estimated. In MCNP 6.2 simulations, the shape of a spent HEPA filter is a rectangular parallelepiped form filled extensively with a pleated filter medium made from a mixture of glass fibers in the metal filter case. The most significant γ -ray-emitting radionuclides are ^{134}Cs (604, 705 keV), ^{137}Cs (662 keV), and, ^{60}Co (1,173, 1,332 keV) in intermediate level radioactive waste (ILW) and its whole size is approximately 57 cm (H) $\times$ 57 cm (D) $\times$ 26.4 cm (H). The simulation results were compared in terms of the working distance, estimated working time, and working posture and demonstrate the precisely estimated radiation exposure dose in the various working posture for separation process of a spent HEPA filters. This study offers guidelines for the radiation safety of pre-treatment workers engaged in the separation process for a spent HEPA filters.

Keywords: Spent HEPA filters, Pre-treatment workers, Collective dose, MCNP 6.2, PIMAL phantom

Preparation of Full System Decontamination of Kori #1 NPP

Hyun-Jun Jo*, Cho-rong Kim, and Ji-hwan Yu

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*johyunjun@khnp.co.kr

The Kori #1 Nuclear Power Plant (NPP) was permanently shut down on June 18, 2017 after being operated for approximately 40 years. Korea Hydro & Nuclear Power (KHNP) has decided the Full System Decontamination (FSD) to reduce the decommissioning worker's radiation dose and radioactive waste generation. Operation procedures and temporary improvement of some facilities and devices have been prepared for conducting the FSD of Kori #1 NPP. Process parameters and facilities was developed in the research project (Development of Full System Decontamination Demonstration Technology for Light-Water Nuclear Power Plant) sponsored by KETEP and carried out by KHNP and KEPCO KPS, etc. Additionally, KHNP and KEPCO KPS plan to use the developed equipment to perform FSD of Kori # 1 NPP. The one decontamination cycle is organized with three steps that are oxidation, reduction, and purification, and total 3 cycles will be carried. The FSD effect will be evaluated using the radiation rate measurement value before and after FSD.

Keywords: Full system decontamination, Kori #1 NPP, System improvement

Design of the Working Space-Type Vacuum Chamber and the Operating Element Equipment of a Vacuum Induction Melting System

Sung-Ryul Kim, Jung-Min Oh, Kyu-Tae Park, Ba-Ro Lee, Chan-Hee Park, Bo Bae Cho, Ji-Ung Kim,

Hye-Jin Kim, Hyun-Young Shin, So-Yun Jeong, Oh-il Kwon, and Wook Jae Yoo*

Orbitech Co., Ltd.

*wjyoo@orbitech.co.kr

This study is about the development of a vacuum induction melting (VIM) system for the treatment of metal waste generated during the operation or decommissioning of nuclear-related facilities, and aims to minimize permanent disposition and maximize self-disposal by melting metal waste in a vacuum atmosphere. From February 2023 to June 2024, this research team advanced the design of the commercial-scale (600 kW, 500 kg/batch) VIM system by changing the vacuum chamber to square box-type (Ver. 1.0), cylindrical-type (Ver. 2.0), and cylindrical separation-type (Ver. 3.2) in consideration of the operational efficiency, safety, and decontamination ease of the VIM system and the improvement of the pre-developed pilot-scale (150 kW, 100 kg/batch) VIM system.

First, square box-type was determined that it would be difficult to manufacture a double jacket of a vacuum chamber for cooling and design cooling water circulation, and then, in the case of a cylindrical-type, it was considered that there were problems with the arrangement of various unit devices located inside the vacuum chamber, decontamination, and difficulty in replacing parts, so it was stopped at the conceptual design stage. In the case of the cylindrical separation-type, the vacuum chamber is separated from the front end where the mold transfer device is located and the rear end where the melting furnace is located, so it is considered to be more advantageous in charging the melting furnace of metal waste, removing slag, and replacing various unit devices located inside the vacuum chamber than the existing types, therefore the basic design was carried out. However, during the final review before the detailed design, it was concluded that the ease of decontamination and parts replacement for contaminated areas of internal unit devices expected to occur during the operation of the VIM system and securing a melting work space inside the vacuum chamber was still insufficient.

Accordingly, from July 2024, this research team came up with the idea of a work space-type (Ver. 4.0) vacuum chamber so that workers can enter the vacuum chamber and perform air induction melting (AIM) work, and started basic design after conceptual design. In addition, as of August 2024, design enhancements are being carried out for operating element equipment attached to the inside/outside of the vacuum chamber to improve the operational efficiency of commercial-scale VIM systems and simplify work procedures. The working-space VIM system under development is expected to be a key facility for the treatment/disposal of metal waste as it has the advantage of protecting workers from molten metal splashes or melting furnace explosions and minimizing the risk of radiation exposure.

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Keywords: Vacuum induction melting system, Working space-type vacuum chamber, Operating element equipment, Metal waste

Improving Cutting Quality Based on Different Cutting Speeds of SS400 Thick Plate by Optimizing Fiber Laser Pulse Parameters

Dong-Hyun Kim*, Moo-Keun Song, Min-Gyu Choi, Yong-Won Ma, and Yong-Guk Son

Korea Research Institute of Decommissioning (KRID)

*dhkim@krid.re.kr

The remote cutting of activated structures is a crucial task in the process of nuclear power plant decommissioning, and it is considered a key technology for minimizing radiation exposure to workers. Remote cutting technologies are generally classified into mechanical, thermal and electrical cutting. Among the various thermal cutting methods, laser cutting is more effective than other techniques in terms of cutting speed (reducing operation time) and secondary waste minimization.

Fiber laser cutting involves irradiating the cutting area with light energy of a specific wavelength to melt the material, followed by the use of cutting gas to remove the molten material. The quality of the cut is influenced by various factors such as stand-off distance, power, cutting speed, and pulse parameters. If the cutting conditions are not optimal, bur or dross may form on the lower part of the cut. These impurities can negatively affect the separation and handling of the cut material, so it is important to minimize them by optimizing the cutting conditions.

In this study, SS400 thick plates, commonly used in internal structures of nuclear power plants, were cut using a fiber laser, and the pulse parameters were optimized to achieve improved cutting quality at different cutting speeds. This optimization improved the cutting quality in acceleration and deceleration zones during both straight and curved cuts. The findings of this study are expected to contribute significantly to high-speed cutting of SS400 thick plates using fiber lasers in the future decommissioning of domestic nuclear power plants.

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Keywords: Fiber laser, Laser cutting, Pulse parameter, Cutting speed, Cutting quality

Development of Real-Time 3D Gamma-Ray Imaging System

Min Chul Kim*, Young Il Na, and Ji Hwan Yu

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kim.minchul@khnp.co.kr

Nuclear dismantling is a top priority to prevent radiation harm to workers and the environment. To this end, it can be said that it is important to evaluate radiation characteristics continuously performed before and during dismantling. Radiological characteristics evaluation can be largely divided into radiochemical analysis through sample collection and field measurement. The pre-dismantling characteristic evaluation requires representativeness of samples and measurements, reliability and reproducibility of analysis results. In overseas dismantling nuclear power plants, gamma-ray visualization equipment is used from the HSA (Historical Site Analysis) stage to the Final Status Survey (FSS). In this study, KHNP's unique equipment is developed for dismantling nuclear power plants to increase utilization in dismantling nuclear power plants.

The real-time 3D gamma-ray imaging system consists of a hardware part including a detector and the like, and a software part including algorithms and programs. The hardware consists of a system housing including a detector sensor, a collimator mask, a fisheye lens, and a near-field lidar and a notebook for remote lidar, equipment control and display. The operating software consists of an image reconstruction algorithm, an image matching algorithm, a radioactivity measurement algorithm, a nuclide analysis algorithm, a dose measurement algorithm, and an integrated control program.

In order to verify the performance of the developed gamma-ray visualization equipment and derive improvement, field tests were conducted on the containment building of Kori Unit 1 and Wolsong Unit 1 and the gamma-ray irradiation facility in the KHNP Central Research Institute. In addition, the planned performance goals such as radioactive contamination measurement accuracy, radioactivity measurement accuracy, dose estimation accuracy, imaging resolution, detector weight reduction, detector and lidar coordinate matching were achieved by an accredited agency.

The equipment developed to contribute to the measurement, visualization, and quantification of radioactive contamination in operating and dismantling nuclear power plants will be continuously supplemented. Through this, it is intended to improve radiation safety management capabilities through rapid measurement and provision of visualization data of radioactive contamination.

Keywords: Real-time 3D gamma-ray imaging system, Visualization, Measurement

A Review of the Condition Assessment of Low-Level Waste Containers at the Low-Level Storage Buildings at Western Waste Management Facility in Canada

Jin-Won Son*, Gang-woo Ryu, and Suk-Bon Yoon

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*son.jinwon@khnp.co.kr

Canada manages its low-level radioactive waste by storing it at the Low-Level Storage Buildings (LLSB) in Nuclear Sustainability Service Western (NSSW). OPG manages radioactive waste and radioactive materials at three NSS sites. NSSW, located in the Bruce region, is a facility that manages low-level and intermediate-level radioactive waste from the Darlington, Pickering and Bruce Nuclear Power Plant.

OPG has established and Aging Management Plan (AMP) that outlines the requirements for monitoring the integrity of containers in the LLBS that are risk of deterioration. As per the AMP, aging monitoring activities were carried out for a sample of containers including external and internal visual assessments, ultrasonic wall thickness measurements, metallographic analyses, and radionuclide analyses of waste sample. The following 5 types of containers were selected as evaluation targets based on AMP criteria: (1) Low level resin containers, (2) Solidified active liquid waste containers, (3) Ash containers, (4) Drum containers, (5) Non-processible containers. The mainly results of the 2019 assessment are as following. (1) The resin box container which showed significant signs of accelerated corrosion with ongoing leaks from the residual water in the resin was moved to other area to contain any further leaks and maintenance repaired the side wall to prevent further leakage. (2) The one cubic flat ash container which was found to be leaking with areas of heavy pitting and corrosion was placed in an overpack. (3) Procedure updates were made to provide guidance on discovery of deteriorated bins during waste handling and storage operations. (4) The ALW containers, the other ash containers, drum container and the non-processible blue bins were all in good condition and show no signs of accelerated corrosion, apart for some surface corrosion in some areas which may be due to scraped paint or abrasion from mechanical handling, and, therefore, are expected to meet their intended design life. (4) Updates to the Aging Management Program were completed following this assessment that prompted revisions to existing AMPs and new AMPs being developed in support of the scoping and screening criteria which outlines the requirements for the Aging Management Program.

The purpose of this paper is to review the results of the assessment of low-level radioactive waste containers stored at the LLSBs of the NSSW, which comprehensively manages Canada's low-level radioactive waste, and to utilize them for domestic decommissioning projects.

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Keywords: Radioactive waste, Low-level radioactive waste, Radioactive waste container, Nuclear power plant decommissioning, Dismantling process

A Review on Overseas NPPs DCGL Evaluation of Buried Piping

Gang Woo Ryu*, Young Il Na, Hyung Woo Seo, and Kwang Ho Jo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*gangwoo.ryu@khnp.co.kr

When decommissioning nuclear power plants, underground piping may remain below ground according to the decommissioning plan. The residual radioactivity of these buried piping can potentially migrate through natural diffusion processes into the aquifer and affect radiation health risks for future users of the site. Additionally, there is also a possibility that some of the buried piping could be excavated during the development of wells for drinking and irrigation purposes, which would also affect radiation health risks. In the case of decommissioning at US nuclear power plants, they have evaluated the residual radioactive concentration induced by remaining buried piping after decommissioning. Therefore, in this study, we analyzed the methodology used to evaluate the Derived Concentration Guideline Levels (DCGL) of buried piping at overseas decommissioning nuclear power plants such as Zion, Rancho Seco.

In the Zion NPP, two scenarios were assumed to derive DCGL for buried piping. The first scenario is that all piping is excavated, and the second scenario is that all piping remains in-situ. The in-situ scenario was divided into two cases; assuming that all piping is located 1 m below the ground surface in the unsaturated zone and located in the saturated zone. The lowest in-situ DCGL from either the 1 m deep unsaturated or saturated scenario was assigned as the in-situ buried piping DCGL. Although unrealistic, for the purpose of the bounding modeling approach used, the dose from the excavated and in-situ scenarios are summed to determine the buried piping DCGL. For all scenarios, the exposure scenario assumed a resident farmer scenario. The DCGLs unit of the buried piping is dpm/100 cm².

In the Rancho Seco NPP, buried piping is assumed to disintegrate instantaneously. The disintegrated media is assumed as soil and the media volume is assumed to be equal to the piping volume. Only external gamma exposure from buried piping is considered.

In this study, we analyzed the methodology used to evaluate the DCGL of buried piping at overseas decommissioning NPPs. The results of this study are expected to be used as basic data for buried piping DCGL for Kori unit 1 and Wolsong unit 1.

Keywords: Decommissioning, DCGL, Buried piping

A Study on the Selection of Site-Specific Input Parameter for Overseas NPPs Soil DCGL

Gang Woo Ryu*, Young Il Na, Hyung Woo Seo, and Kwang Ho Jo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*gangwoo.ryu@khnp.co.kr

Kori unit 1 and Wolsong unit 1 permanent shutdown in 2017, 2019, respectively. In Korea, decommissioning of nuclear power plant (NPP) can be completed if the results of Final Status Survey (FSS) are less than Derived Concentration Guideline Level (DCGL). According to NSSC Notice 2021-11, In the event that the licensee employs mathematical prediction models to assess the exposure dose due to residual radioactivity; DCGL evaluation, the licensee shall use characteristics of the facility and site shall be appropriately taken into account in the assessment. It is important to reflect the site specific characteristics. Therefore, in this study, we analyzed the method of selecting site specific input parameter at overseas decommissioning nuclear power plants such as Humboldt Bay, Yankee Rowe.

In the Humboldt Bay NPP, site specific characteristics were reflected for 7 input parameters; Area of the Contaminated Zone, Length Parallel to Aquifer Flow, Contaminated Zone Erosion Rate, Soil Type, Field Capacity, Irrigation Rate, Well Pumping Rate. In the Yankee Rowe NPP, site specific characteristics were reflected for 10 input parameters; Area of the Contaminated Zone, Contaminated Zone Erosion Rate, Humidity in Air, Average Annual Wind Speed, Precipitation, Irrigation Rate, Field Capacity, Saturated Zone Hydraulic Gradient, Well Pumping Rate, Watershed for Nearby Stream of Pond. Most of the other input parameters used NUREG/CR-6697 and RESRAD defaults.

In this study, we analyzed the method of selecting site specific input parameter at overseas decommissioning NPPs. The results of this study are expected to be used as basic data for DCGL evaluation.

Keywords: Decommissioning, Site-specific, DCGL, Parameter

A Review of the L&ILW Criteria and Management Plan in Canada From Existing International Criteria and Experience

Jin-Won Son*, Gang-woo Ryu, and Kyung-ho Roh

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*son.jinwon@khnp.co.kr

The Nuclear Waste Management organization (NWMO) has submitted its recommendations for an integrated radioactive waste management strategy for 2023 to the Ministry of Energy and Natural Resources in Canada. Canada has some long-term disposal plans in place for its radioactive waste, but it does not yet have a radioactive waste disposal facility, and radioactive waste is currently stored in licensed temporary facilities. For this reason, in November 2020, the Canadian government requested the NWMO to establish an integrated radioactive waste management strategy for Canada, and the recommendations made in July 2023 should be reviewed and utilized for domestic radioactive waste management and decommissioning projects such as those of Kori Unit 1 and Wolsong Unit 1.

The main contents of the principles for implementing the newly established Canadian Integrated Radioactive Waste Management Strategy are as follows. (1) Two key recommendations are presented to address the gap in long-term radioactive waste disposal plans: (a) Intermediate-level, non-nuclear spent fuel and high-level radioactive waste will be disposed of in a deep repository to be promoted by the NWMO. (b) Low-level waste is disposed of in a multiple near surface disposal facilities managed by the waste generator and owner. (2) The integrated management strategy includes four principles for effective implementation of the recommendations: (a) The consent of the local communities and Indigenous peoples. (b) The design of facilities should prioritize the protection of water. (c) Long-term caretaking should be established for disposal facilities. (d) Need to take actions now not defer to future generations. So, in Canada, the NWMO, which is the Canadian Radioactive Waste Management Authority, manages high-level and intermediate-level waste, while low-level waste is managed by utilities themselves.

Also, in Canada, radioactive waste is classified into three categories: high-level, intermediate-level and low-level. However, the standards for distinguishing between intermediate-level and low-level radioactive waste are known to be nuclear and inadequate. So, in this paper, we will organize and compare the classification standards for radioactive waste in major countries such as USA, France and South Korea including IAEA.

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This work was prepared as part of a report to attend the 2024 Canadian Nuclear Society (AnnCon2024, held from 16th to 20th June).

Keywords: Radioactive waste, Intermediate-level radioactive waste, Low-level radioactive waste, Radioactive waste container, Nuclear power plant decommissioning, Dismantling process

A Review on the Process and Construction of Nuclear Power Plants Decommissioning in Domestic and Overseas

Suk-Won Jung*, Suk-Bon Yoon, and Hyung-Woo Seo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sukwon11@khnp.co.kr

Currently, the nuclear decommissioning market is expected to be 123 trillion won by 2030, and if the entire world's operating Nuclear Power Plants (NPPs) is decommissioned within the next 50 years, a market of more than 400 trillion won is expected to open. In Korea, the permanent suspension of Kori unit 1 (June 2017) and Wolsong unit 1 (December 2019) have been suspended permanently, and the approval of the Final Decommissioning Plan (FDP) for Kori unit 1 is currently in progress according to related laws and regulations. In addition, the Wolsong unit 1 FDP is currently in the process of reviewing the suitability of documents after submitting it to the regulatory agency (KINS). In this regard, the guidelines for reviewing the FDP require standards such as technical capabilities and decommissioning plans, and the contents of the exposure radiation generated during the decommissioning process. Therefore, this study, we reviewed the contents of the construction and process during the decommissioning of NPPs.

The design of construction and processes in the event of decommissioning NPPs is a very important step in terms of the decommissioning schedule and cost reduction. The technology to be used in the process of decommissioning NPPs plays an important role in terms of the safety of the construction, and the process considering the decommissioning schedule is directly correlated with the cost of the construction. Therefore, this study attempted to analyze the cases of decommissioning domestic NPPs related to construction and processes and to compare them with cases of decommissioning overseas NPPs such as the United States, Japan, and Europe. In the case of Korea, there was no reason to decommissioning commercial NPPs, so the case of decommissioning research NPPs was reviewed, compared and reviewed with overseas cases. In particular, it was compared and reviewed in terms of technical approach and process efficiency, and systematic management of the latest technologies and processes applied abroad was reviewed.

When decommissioning a NPPs, how the process is designed is expected to have a significant impact on the cost of decommissioning NPPs and the construction schedule. In the future, if domestic conditions related to construction and process are prepared, it is expected that this study will be helpful as a reference for domestic NPPs decommissioning construction and process.

Keywords: Decommissioning, Nuclear power plants, NPPs, FDP, Process and construction

Radiological Characteristics of Structure and Concrete of the Decommissioning Nuclear Power Plant

Hyuchang Choi*, Jihwan Yu, and Hyunmin Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hyuchang.choi@khnp.co.kr

The evaluation of radiological characteristics of nuclear power plant (NPP) is important because it impacts the entire process from the decommissioning planning stage to site restoration. In particular, the characterization survey performed before decommissioning NPP can be extensively utilized in various aspects, including the scale of decontamination, the selection of decommissioning methods, etc.

In this study, the radiological characteristics of structure and concrete was performed on the surface layer of contaminated concrete and structures of reactor and service building to determine the characteristics of the main structure at of CANDU type NPP. Samples from reactor building and service building were collected, based on the surface contamination.

The results showed that both in the reactor building and service building, main gamma nuclides were Co-60 and Cs-137. It will be used as the basic data for classifying radioactive waste grade and determining the amount of waste of structure and concrete for decommissioning NPP.

Keywords: Decommissioning, Radiological characteristics, CANDU, Decontamination

Radiological Characteristics of Systems of the Decommissioning Nuclear Power Plant

Hyuchang Choi*, Jihwan Yu, and Hyunmin Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hyuchang.choi@khnp.co.kr

The decommissioning of nuclear power plants (NPPs) is a complex and multifaceted process that involves the safe dismantling of nuclear facilities and the management of radioactive materials. One critical aspect of this process is the radiological characterization of system and component, which is essential for ensuring safety, compliance with regulatory requirements, and the protection of workers and the environment. Generally, radiological characterization involves the assessment of radioactive contamination levels in equipment of facility.

In this study, samples were taken for four systems including the Shutdown Cooling System at CANDU type NPP and analyzed fourteen gamma nuclides to determine radioactive contamination. Although there were differences depending on system, the main pollutants were Co-60 and Sb-125. This result will be used as the basic data for classifying radioactive waste grade and determining the amount of waste for decommissioning NPP

Keywords: Decommissioning, Radiological characteristics, CANDU, Characterization

Preliminary Study on the Evaluation of Radioactive Contamination in Concrete Structures Generated During the Decommissioning of Nuclear Power Plants

Seongjin Maeng^{1,*}, Seyoung Yu², Sang Hoon Lee², Sang June Park¹, and Minji Park¹

¹Korea Hydro & Nuclear Power (KHNP)

²Kyungpook National University

*sjmaeng2319@khnp.co.kr

The radioactive concrete expected to be generated during the decommissioning of nuclear power plants and nuclear facilities must be assessed for radioactive contamination to reduce the amount of radioactive waste. It is essential to distinguish between radioactive and non-radioactive waste. In response to this need, research on decontamination methods to minimize the generation of radioactive waste is actively being conducted.

In this study, the position and radiation were estimated to identify radioactive contamination in concrete structures during the decommissioning of nuclear power plants and related facilities in order to reduce the generation of radioactive waste. Specifically, MCNP simulations were conducted using gamma-ray detectors on concrete samples, assuming contamination with ¹³⁷Cs and ⁶⁰Co. Future study will compare actual measurement results with the simulation data to validate the theoretical evaluation method.

Keywords: NDM, MCNP, Gamma-ray detector, Building materials

Comparison of FDP Dose Evaluation Input Data for Kori-1 and Wolsung-1

Young-Il Na* and Hyung-Woo Seo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*eaglewater012@khnp.co.kr

Domestic nuclear power plants such as Gori Unit 1 and Wolsung Units 1 and 2 are preparing for decommissioning. The Gori Unit 1 is undergoing licensing review of its dismantling plan, while the Wolsung Unit 1 has entered the public consultation phase for its dismantling plan. These reactors have been in normal operation for at least 30 years, and their decommissioning will require significant costs and time. Among these costs, worker radiation protection must be considered. Therefore, this paper aims to discuss the points of comparison between the measures taken for worker radiation management during the decommissioning process of Gori Unit 1 and Wolsung Units 1.

There are two types of nuclear power plants worldwide: light water reactors and heavy water reactors. Before dismantling begins in practice, an ALARA (As Low as Reasonably Achievable) plan should be established for decommissioning and a high-radiation expected reactor decommissioning process should be established. In the case of Gori Unit 1 and Wolsung Unit 1, worker dose limit evaluations are being conducted through simulation computational simulations and interpretations. Additionally, studies to verify actual occupancy time of workers in management areas (called occupied individuals), respiratory rates of workers, and difficulty assessment of work tasks are currently underway.

Nuclear power plants emphasize safety above all else. The predicted radiation doses for workers at the Gori 1 and Wolsung 1 reactors currently being decommissioned must be calculated. In this study, we aim to understand the differences in worker dose assessments between Gori 1 and Wolsung 1 units as well as any process-specific differences. This information can serve as a basis for evaluating whether appropriate measures have been established to maintain ALARA and ensure compliance with occupational dose limits until the completion of decommissioning.

Keywords: Decommissioning nuclear, Dose evaluation, FDP

Review of Plant Design for Preventing Leakage From Radioactive Materials

Young-Il Na* and Hyung-Woo Seo

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*eaglewater012@khnp.co.kr

Recently, power plants have been designed and constructed according to the latest technological and development standards since their inception. In particular, for nuclear power plants, leakage of radioactive materials can cause social problems, so design concepts that shield or reduce radioactive materials are incorporated. These design concepts play an important role during operation as well as decommissioning of the plant, which is why the plant must be continuously managed and improved while operating.

In US nuclear power plants, they follow Reg Guide 8.8 regarding the ALARA principle. This includes guidance on everything from introducing the concept of ALARA to its background and design characteristics for facilities that adhere to the ALARA principle. It also requires establishing programs and procedures to meet US regulatory requirements, as well as providing minimum design criteria. These design items are listed below.

For nuclear power plants currently under construction, we create drawings and models of piping embedded in walls to visualize what is hidden behind them. This provides workers and operators with valuable information and convenience, as well as monitoring for leaks. In Korea, these design details must be recorded in decommissioning plans. They describe changes made during design and operation, along with management measures for those changes. The facilities have been installed according to designs that incorporate the following aspects:

We have analyzed US Reg Guide 8.8 which has helped us in developing our guidelines and incorporating them into the design of our power plants. These designs must be maintained throughout their entire life cycle, from construction to decommissioning. Additionally, they need to be continuously monitored for validity. Radiation safety is an essential aspect that needs to be ensured at nuclear power plants. By preventing radioactive material leakage incidents, which can cause many social issues, and ensuring that no leaks occur even if there were any accidents due to the design itself, the public acceptance of nuclear power plants increases significantly.

Keywords: Decommissioning nuclear, Decommissioning feasibility designs

Fixation of Decommissioning Waste and Measurement Method of Structural Stability at KRR 1&2

Junhyuck Im*, Dooseong Hwang, Junhee Lee, Minyoung Kang, and Geun-Ho Kim

Korea Atomic Energy Research Institute (KAERI)

*jhim@kaeri.re.kr

The decommissioning of Korea Research Reactor Units 1 and 2 (KRR-1&2), the first research reactors in South Korea, began in 1997 and the decommissioning status is currently proceeding with phase 3. It is essential to dispose of all decommissioning waste stored at KRR-1&2 site prior to final radioactivity survey and site restoration. Approximately 40 drums of concrete- and soil-powder waste and other dust waste classified as dispersible materials are stored within KRR-1&2 site. These types of wastes can only be disposed of through fixation (or immobilization) as waste acceptance criteria.

There are many factors to consider when fixing dispersible waste, and the selection of a fixation treatment method is one of the most important factors because fixation method can directly affect the increase in waste volume. According to the waste disposal case in 2015 from KRR-1&2 site, Portland cement was used to fix the dispersible wastes, and the fixation treatment was performed at a weight ratio of dispersible waste : cement : water of approximately 37:30:33, which resulted in an increase in the volume of waste by approximately 300%. In this fixation process, a mixing ratio test for Portland cement will be conducted first, but a test on a fixation method using non-shrinkage grout will also be conducted separately to minimize the increase in waste volume.

In order to verify the structural stability of fixed waste, the compressive strength of the fixed waste must be 3.44 MPa or more, and conventional methods such as uniaxial compressive strength measurement have been widely applied to verify the compressive strength. However, the conventional method has the problem that dispersive wastes are regenerated because the sample is destroyed, and there may be concerns about the spread of contamination during transport of the sample for measuring the uniaxial compressive strength. Therefore, we plan to introduce statistical techniques using ultrasonic velocity measurement methods commonly used in general construction industry sites to confirm the possibility of applying them to characterization of actual radioactive waste.

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Keywords: Decommissioning waste, Fixation (or immobilization), Concrete powder, Soil powder, Waste acceptance criteria

A Study on the Application of Electrolyte Regeneration-based Electrochemical Decontamination Technology for Clearance of Radioactive Metallic Waste Generated During NPPs Decommissioning

Jun-Yeol An*, June-Hyun Kim, and Saebinna Lee

Korea Research Institute of Decommissioning (KRID)

*ajy0314@krid.re.kr

The amount of decommissioning waste generated during nuclear power plant decommissioning is expected to be approximately 6,200 tons, of which the total of radioactive metals, contaminated metals, and contaminated metal plates is 4,300 tons, accounting for approximately 70% of the total waste. At this time, if an appropriate decontamination method is applied to lower the level to the clearance level, the effect of reducing decommissioning/disposal costs due to waste reduction can be achieved. Therefore, this study aims to develop an electrolytic decontamination technology for clearance of radioactive metallic waste.

Electrochemical decontamination is a mtype of chemical decontamination that uses the principle of electrolysis, where the decontaminated material becomes the anode and the conductor becomes the cathode. Then, a current is passed between the two electrodes to remove the corrosion oxide film and the surface layer of the base material on the metal surface through an oxidation reaction. If an acid is used as an electrolyte during electrochemical decontamination, the electrode must be made of a material with excellent corrosion resistance, and there is a safety risk due to leakage of the acid solution. However, if a neutral salt is used, it is easy to handle, and there is no need to use a material with excellent corrosion resistance for the electrolysis device and equipment that comes into direct contact with the electrolyte solution, so it can also achieve a cost-saving effect.

In this study, complex-shaped small-sized metal waste was selected as the target waste, and Co-60 was selected as the main target nuclide based on previous cases related to radioactive metallic waste. The scale of the equipment developed through the existing electrochemical decontamination research is pilot-scale, but the equipment developed through this study is a TRL 9 demonstration scale aimed at actual application in the decommissioning site. In particular, this equipment aims to minimize secondary waste generation and maximize electrolyte regeneration efficiency and safety by using neutral salt. If this technology is applied to actual decommissioning sites in the future, it is expected to contribute to reducing decommissioning costs and solving the problem of early saturation of the disposal site.

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Keywords: Decommissioning, Electrochemical decontamination, Radioactive metallic waste, Clearance

Evaluation of Calibration Factor for Scan Survey to Identify Area of Elevated Activity in Various Detection Conditions

Sang Bum Hong*, Su Jung Min, and Bum Kyoung Seo

Korea Atomic Energy Research Institute (KAERI)

*sbhong@kaeri.re.kr

The release of the site and building from regulatory control is the final stage of the decommissioning project. The planning, implementation, and assessment for the final status survey of the site and building of the KRR 1&2 has been conducted to demonstrate that residual contamination is lower than the release criteria. MARSSIM provides information on planning, conducting, evaluating, and documenting environmental radiological surveys of surface soil and building surfaces for demonstrating compliance with regulations. Direct measurement and scan survey for surface contamination are generally performed by placing the detector on the surface or on the specific surface using portable detector. Scan surveys are determined during survey planning and implementation for final status survey to identify area of elevated activity. The small areas of elevated activity typically call as a hot spots presented or detected of the survey unit in the building. Scan surveys are usually applied quick and inexpensive to preform than sample analysis relatively. Scan surveys of beta and gamma contamination level measure by gas preoperational counter of plastic scintillator normally used. Measurement condition for detector is held around 1~2 cm from the surface and moved at constant speed to detected the investigation level. The measurement conditions such as distance from the surface and scan speeds in important to identify the hot spots and contamination level. In this study carried out a contamination levels measured for various measurement conditions by change the scan survey distance and speed using portable gas proportional detectors. The calibration factors for different scan survey conditions were evaluated for the decontamination efficiency and residual radioactivity evaluation in a decommission project

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Keywords: MARSSIM, Surface contamination, Scan survey, Measurement condition, Calibration

Dose Assessment Considerations for Subsurface Contamination for Decommissioning Site Release

Sang Bum Hong*, Su Jung Min, Geun Ho Kim, and Bum Kyoung Seo

Korea Atomic Energy Research Institute (KAERI)

*sbhong@kaeri.re.kr

RESRAD is a computer model designed to estimate radiation doses and risks from residual radioactive material, which was widely used for radiological dose assessment for site release. The alternative conceptual models may occur at decommissioning sites such as a contaminated subsurface structures or basements which are backfilled and left as part of the end state of decommissioning. There are many underground structures in the reactor buildings of the KRR 1&2, and such a safety assessment is required for site release of the site and structures. RESRAD code can be conservatively used for assessing contaminant release from subsurface structure located in the unsaturation zone. This study carried out dose evaluation using mass balance model with non-dispersion model with considering dilution factor for subsurface structure. The results of the dose evaluations show that non-dispersion model may result in overestimating the amount of dilution factor with source from subsurface structure. This result can be useful for final status survey and site release of KRR decommissioning site.

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Keywords: Decommissioning, Site release, Underground structure, RESRAD

Development of a Methodology for Verifying the Plant Inventory Database- Quantity Evaluation

~~Jin Yoo^{1,*}, ChangSok Cho¹, Seonggon Kim¹, Gyumin Bae¹, Sanggyoon Chang², and Jongseo Choi³~~

~~¹Nucl. Eng. Serv. & Solut.~~

~~²KEPCO Engineering & Construction Company (KEPCO E&C)~~

~~³DAEWOO Engineering & Construction Co., Ltd.~~

~~*clickjin100@ness.re.kr~~

Withdrawal

Design of a CECE System and Performance Analysis of Hydrophobic Catalysts for Light Water Detritiation

Chan Woo Park*, Daeuk Kang, Hyung-Ju Kim, Hee-Man Yang, and Young-Ho Shin

Korea Atomic Energy Research Institute (KAERI)

*chanwoo@kaeri.re.kr

The combined electrolysis and catalytic exchange (CECE) process, which integrates water electrolysis and liquid-phase catalytic exchange, can achieve a high total separation factor and is more compact compared to other processes. For the development of CECE technology aimed at treating tritiated water, it is crucial to analyze the performance of wet-proofed catalysts, which in turn informs the design of CECE catalyst towers and related systems. In this study, we calculated the overall mass transfer coefficient of a hydrophobic catalyst at a lab-scale using a small column. Based on these results, a CECE system for the hypothetical treatment of tritiated water was designed. This enabled the determination of the size of the catalyst tower and electrolyzer system.

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Keywords: Tritium, Deuterium, Separation, LPCE, Catalyst, CECE

Performance Evaluation of Wet-Proofed Catalysts for Liquid Phase Catalytic Exchange (LPCE)

Chan Woo Park*, Hyung-Ju Kim, Hee-Man Yang, Young-Ho Shin, Jong-Yun Kim, and Sang Ho Lim

Korea Atomic Energy Research Institute (KAERI)

*chanwoo@kaeri.re.kr

Liquid phase catalytic exchange (LPCE) is a technology with a high separation factor for hydrogen isotope separation. While LPCE is applied to tritium extraction, it can also be used for deuterium separation from natural water to produce heavy water. Evaluating the performance of wet-proofed catalysts is critical for analyzing hydrogen isotope separation performance and process design in LPCE-based systems for heavy water production. In this study, we designed and fabricated an evaluation device to analyze the mass transfer coefficient, pressure drop, and other characteristics of the catalyst. Using this device, we further examined the water-vapor exchange reactions and the water-hydrogen isotope exchange characteristics of the catalysts.

Acknowledgements

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Keywords: Deuterium, Separation, LPCE, Catalyst

Case Study of Residual Radioactivity Assessment for Sites Release

SuJung Min, GeunHo Kim, BumKyoung Seo, and SangBum Hong*

Korea Atomic Energy Research Institute (KAERI)

*sbhong@kaeri.re.kr

In this paper, we analyzed the DCGL setting method and measurement method of overseas nuclear power plants(NPPs) and reviewed the application method to domestic NPPs. The safety of decommissioning site release of a nuclear facility can be proven by satisfying the site release criteria for the residual radioactivity level of the site. In the United States, the unlimited decommissioning standard must meet 250 uSv/y, with a focus on doses to soil, groundwater. In particular, for the Rancho Seco Nuclear Power Plant, RESRAD version 6.3 was used for site-specific dose modeling to consider the impact on the subsurface soil, and derived concentration standards (DCGL) for each radionuclide were calculated based on the analysis results of soil sampling. In the case of Rancho Seco, subsurface soil contamination was analyzed in an site reuse strategy for final goal as industrial site. The subsurface soil is a soil layer that is more than 15 cm deep from the surface soil. Contaminated groundwater within the site increases the complexity of dose modeling and has a significant impact on the setting of site release criteria. Realistic dose modeling including basement fill models such as subsurface soil can significantly reduce the amount of decontamination required to meet site release criteria. A sufficient investigation of the radiological characteristics of the site is necessary, including an investigation of subsurface contamination, which is a limitation of MARSSIM. Overseas cases from the United States, Germany, and Japan show that setting specific goals for site release, systematic regulations, and measurement/analysis technologies to satisfy regulations are important factors to consider in site restoration.

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Keywords: DCGL, MARSSIM, Site release, Subsurface soil, RESRAD

Stabilization of Residues From the Carbonization Process Using Glass Composite Materials

Jaewoong Hwang^{1,2}, Jaseung Koo¹, and Keunyoung Lee^{2,*}

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*lky@kaeri.re.kr

The elements generated through the carbonization process, one of the incineration replacement technologies for treating decontamination paper, include Ca, Mg, and Al which are captured in water in the forms of CaCl_2 , MgCl_2 , and AlCl_3 through a chlorination process. Zeolite, a representative inorganic adsorbent, was selected for the coagulation and precipitation of these residues. Furthermore, a new adsorbent, MFC-zeolite, was synthesized by impregnating zeolite with MFC (Metal ferrocyanide), which is known for its excellent cesium adsorption properties, into the zeolite. Various types of zeolites were impregnated with MFC, and 4A-type zeolite, which was confirmed by EDS to have the highest Co and Fe content, was selected as the adsorption and precipitation agent.

The precipitate formed using MFC-4A was dried, ground to a particle size of less than 200 μm , and then compressed at 60 MPa using a hydraulic press to produce green bodies. The green body is 13 mm in diameter and about 10 mm in height. These green bodies were sintered in an electronic muffle furnace at 750°C and 850°C, achieving volume reduction rates of 41% and 45%, and compressive strengths of 2.68 MPa and 6.56 MPa, respectively. To lower the sintering temperature while maintaining stable compressive strength, B_2O_3 was added to the dried precipitate powder as an additive, in amounts ranging from 5 to 20wt%. Sintering was then conducted at temperatures between 450°C and 850°C. With only a small amount of B_2O_3 added, the compressive strength exceeded the disposal standard of 3.44 MPa under all temperature conditions. The maximum volume reduction and compressive strength of the sintered body, observed at 67% and 116 MPa, respectively, were achieved at 850°C with a 10wt% B_2O_3 content. Mineralogical changes before and after sintering were analyzed using XRD, and microstructural changes were observed with SEM.

The results of this study demonstrated that the production and evaluation of glass-ceramic sintered bodies for the final disposal of the precipitate achieved high levels of physical stability and volume reduction.

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Keywords: Glass composite material, Sintering, Stabilization, MFC-4A, Volume reduction

MCNP Model for Estimation of Radioactivity Source Term in a Pressurized Heavy Water Reactor System

Yong Nam Kim¹, Sanghwa Shin^{1,*}, Chang-Lak Kim¹, and Do Hyun Kim²

¹Kepeco International Nuclear Graduate School (KINGS)

²Korea institute of Fusion Energy (KFE)

*sshwa73@naver.com

On December 24, 2019, Wolsong Unit 1 was permanently shut down, which is the first CANDU type reactor in Korea and the decommissioning/dismantling process is in progress. Additionally, three heavy water reactors, including Wolsong Units 2-4, are operating in Korea. It is necessary to establish a plan for decommissioning/dismantling of these heavy water reactor systems in the future. To ensure the safety of workers in the field of radiation exposure, a work process is established based on radioactivity evaluation for the reactor components.

The radionuclide inventory calculation is highly dependent on the neutron energy spectrum as well as the magnitude of the neutron flux. From the sense of conservative perspective on the safety of workers, selecting the highest neutron flux and the softest energy spectrum is desirable to prevent the risk of underestimating worker exposure. On the other hand, if the radionuclide source term is overestimated, the radioactive waste yield increases, which leads to an increase in the management burden. Therefore, from the viewpoint of optimization in terms of trade-off between the worker safety and economic efficiency, an excessively conservative approach is not desirable. In other words, an optimized calculation model is required that can obtain neutron flux distribution and neutron energy spectrum as close to reality as possible.

In this study, we discuss Monte Carlo modeling techniques for radioactivity evaluation for heavy water reactors to establish a realistic calculation system in the sense of the above-mentioned optimization perspective. Firstly, we calculate the neutron flux and neutron energy spectra along with the location of the nuclear fuel channel in the calandria tank of Wolsong Unit 1 reactor. Sensitivity analyses is carried out in terms of the variation of neutron flux and the change in neutron energy spectrum along with the radial distance of the nuclear fuel channel from the calandria center. Next, the radionuclide inventory in the pressure tube and the calandria tube of the nuclear fuel bundle at a specific location is estimated by solving the radioactivity equation. The MCNP 6 and the FISPECT-II code are used as a neutron transport calculation tool and radioactivity calculations, respectively. Ultimately, according to the location inside the calandria tank, nuclear fuel channels and fuel bundles are classified into groups with the similar level of neutron importance to obtain an optimized MCNP model for activation calculation of CANDU type reactor system.

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Keywords: CANDU reactor, Decommissioning/dismantling, Radioactivity source term, Activation calculation, MCNP, FISPECT-II

Three Phase AC (Alternative Current) Arc Plasma Torch System for Treatment of Liquid Radioactive Wastes

Dong-Hyun Lee, Nam-Ki Lee, Hae-Won Park, Sang-Yun Paek, Chang-Woo Park, Ye-Gyeom Lee, and Jun-Ho Seo*

JeonBuk National University (JNBU)

*jhseo@jbnu.ac.kr

Recently, thermal plasma systems have attracted many attentions for treatment of LILW (Low and Intermediate Level Waste). In this work, a three phase AC (Alternative Current) arc plasma torch system is introduced as a new heat source for radioactive waste treatment, in particular including liquid waste. Since three phase AC (Alternative Current) arc plasma torch consists of three carbon electrode, which are symmetrically cined to each other with a converging point, wastes in the form of powder or liquid can be injected along the centerline of plasma flame that is formed alternately between electrode tips. Thanks to this structural features, this type of plasma torch is expected to give a efficient heating route to the injected waste powder and liquid droplets, resulting in rapid thermal decomposition of organic materials inside solid wastes or easy evaporation of water in liquid wastes. In order to obtain basic performance data for this type of plasma torch system, we have designed a cylindrical graphite chamber confining three carbon electrodes, and measured temperatures and velocities of plasma flame exiting this graphite chamber by using an enthalpy probe. Detailed experimental results will be presented in poster session together with some discussion on the application of AC plasma torch system to treatment of specialty LLIW including liquids.

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Keywords: AC (Alternative Current) arc plasma, Liquid radioactive waste, Plasma characteristics, Enthalpy probe

Effect of Seawater on Distribution Coefficient and Interaction Cesium and Strontium With Clay Minerals

In-Ho Yoon*, Ilgook Kim, Youngho Sihm, Geun-Ho Kim, KwangSeong Jeong, and Chan Woo Park

Korea Atomic Energy Research Institute (KAERI)

*ihyoon@kaeri.re.kr

The behavior of radionuclides in coastal areas near nuclear facilities is significantly complicated by seawater intrusion. This study presents a comprehensive assessment of the effects of seawater intrusion on the mobility of cesium (Cs) and strontium (Sr), with a particular focus on their interactions with various clay minerals including kaolinite, illite, hydrobiotite, and montmorillonite. We employed a combination of batch adsorption experiments, and X-ray diffraction (XRD) analysis to investigate these interactions under diverse seawater intrusion conditions. Our findings reveal that the increase in ionic strength due to seawater intrusion substantially reduces the distribution coefficients (K_d) for both Cs and Sr across all studied clay minerals. The adsorption behavior of these Cs and Sr showed distinct variations across different clay types when subjected to pH changes ranging from 1 to 10. Notably, the presence of major cations in seawater, particularly K^+ and Ca^{2+} , significantly impacted Cs and Sr adsorption through competitive interactions for binding sites. The competitive effect was especially pronounced for Cs in the presence of K^+ ions, likely due to their similar ionic radii and chemical behavior. XRD analyses demonstrated structural changes in some clay minerals under seawater conditions, including alterations in interlayer spacing. These mineralogical changes, coupled with the chemical effects of seawater intrusion, create a complex environment that generally enhances Cs and Sr mobility. Our research highlights that the combination of increased ionic strength, presence of competing ions, and structural modifications of clay minerals acts as key factors in increasing the mobility of Cs and Sr in seawater-intruded coastal environments. These results have important implications for risk assessment and management strategies at coastal nuclear facilities. By elucidating the mechanisms through which seawater intrusion affects Cs and Sr behavior, our study provides crucial insights for predicting contaminant transport in dynamic coastal systems. Furthermore, this work contributes to the broader understanding of Cs and Sr-mineral interactions under variable geochemical conditions, which is essential for developing effective remediation strategies and improving long-term safety assessments of nuclear waste disposal in coastal regions.

Acknowledgements

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Keywords: Radionuclides, Seawater intrusion, Clay minerals, Adsorption, Distribution coefficient

A Review on Example of SSC Recategorization Process for Decommissioning of Nuclear Power Plants

Jihwan Yu*, Hyunjun Jo, Minchul Kim, Hyuchang Choi, and Sukwon Jung
Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)
*liu20893283@khnp.co.kr

The early decommissioning phase of a nuclear power plant is an operation period that requires detailed planning, system modifications, and strategic reconfiguration to facilitate an efficient and safe transition from active operations to full dismantlement. During this phase, the recategorization of systems, structures, and components (SSCs) becomes essential to optimize the decommissioning process. Recategorization helps to identify systems that are no longer necessary for nuclear safety, allowing them to be downgraded or decommissioned. This strategic approach significantly reduces regulatory burdens, maintenance demands, and operational costs, while accelerating the decommissioning timeline.

Evaluating SSCs involves a comprehensive assessment of their role in the plant's defueled state. Systems that were once critical for maintaining reactor operations, such as spent fuel pool cooling systems and reactor safety equipment, are examined to determine whether they remain essential. In many cases, these systems no longer need to meet the same safety-related standards. Downgrading or decommissioning them not only streamlines the decommissioning process but also reduces regulatory oversight and associated costs. Systems that no longer contribute to safety can be dismantled early, which contributes to a faster and more efficient decommissioning process overall.

A key aspect of this phase is the handling and safety of spent fuel. As spent fuel cools over time, the heat load decreases, allowing for modifications to the spent fuel pool systems. These modifications typically involve downsizing cooling systems, which can then be removed once the spent fuel has been transferred to dry storage or offsite. By isolating the spent fuel pool in a "spent fuel pool island", other areas of the plant can be dismantled without affecting the safety of the stored fuel. This approach ensures that the plant can continue decommissioning activities without delays, while maintaining the necessary precautions for spent fuel management.

In addition to the technical modifications, documentation and regulatory compliance are critical throughout the decommissioning period. Regulatory bodies must approve all changes to SSCs, including the downgrading of safety classifications, to ensure that safety is maintained. This involves revising safety analysis reports and technical specifications to reflect the plant's new defueled status. Maintaining compliance with regulatory requirements ensures that the plant continues to operate safely during decommissioning and reduces the risk of unforeseen issues.

This early phase of decommissioning sets the stage for the successful dismantlement of the plant. By effectively recategorizing SSCs and implementing system modifications, nuclear power plants can significantly reduce costs, simplify operations, and expedite the overall decommissioning process. These strategic actions ensure that the transition from active operations to dismantlement is handled safely and efficiently, ultimately leading to a smoother path toward site closure and license termination.

Keywords: Characterization, Autonomous system, Decommissioning, Final status survey

A Study on Groundwater Site Conceptual Model Template for Nuclear Power Plants

Jihwan Yu*, Hyunjun Jo, Minchul Kim, Hyuchang Choi, and Sukwon Jung
Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)
*liu20893283@khnp.co.kr

Groundwater protection plays a crucial role in maintaining environmental safety at nuclear power plants, where the potential for radionuclide contamination poses significant risks. This study presents a detailed and structured framework for developing a robust Site Conceptual Model (SCM), specifically tailored for nuclear power plant sites. The SCM serves as a key tool to guide groundwater monitoring programs, assess risks associated with plant-related contaminants, and understand pathways for potential contaminant migration. It integrates hydrogeological information with data on site-specific sources of contamination, such as historical spills, leaks, and current radionuclide transport mechanisms.

The SCM's development involves comprehensive sections, including an analysis of site geology, hydrology, and hydrogeology, which together form a basis for understanding groundwater flow and contaminant behavior. Also, this emphasizes the importance of identifying both horizontal and vertical groundwater flow patterns, as well as determining the sources and transport of plant-related radionuclides. In addition, it highlights the critical need to regularly update the SCM as new data is collected, such as changes in groundwater elevation, new well installations, or operational changes at the plant that might alter the groundwater flow or increase risks to environmental receptors.

Furthermore, the SCM is designed not only as a technical model but also as a tool for communication. It enables nuclear plant personnel and external stakeholders, such as regulatory bodies, to easily understand and assess the site's groundwater conditions and risks. This is achieved by supplementing written analysis with illustrative graphics, including maps and cross-sections that visually convey key aspects of the site's layout, groundwater flow, and potential pathways for contamination. These cases are recommending best practices for ensuring the SCM remains a reliable and effective tool, such as conducting periodic reviews and aligning the model with the most recent data trends and changes in site conditions.

The lessons learned from previous SCM implementations underscore the importance of using the model as a living document that evolves alongside the site's operational and environmental changes. This proactive approach ensures that the nuclear power plant's groundwater protection program is both technically sound and adaptable, allowing for timely responses to any new potential contaminant release or environmental change. For instance, changes in site pumping activities, the installation of new dewatering systems, or significant modifications to site infrastructure, such as the addition of a spent fuel storage facility, are factors that should trigger immediate reviews of the SCM to maintain its accuracy and reliability.

In conclusion, the SCM framework outlined in this study provides nuclear power plant operators with a practical, detailed, and adaptable approach to managing groundwater protection. By regularly updating the SCM and incorporating best practices, plant operators can effectively mitigate the risks associated with groundwater contamination, ensure compliance with regulatory requirements, and protect both the environment and public health. The continued refinement of the SCM, along with clear communication and graphical representation of site data, will help to ensure that stakeholders are well-informed, and that the plant's groundwater protection program remains resilient and effective over the long term.

Keywords: Site Conceptual Model (SCM), Groundwater, Decommissioning, Release

An Analysis on Design of an Autonomous System for Radiological Characterization

Jihwan Yu*, Hyunjun Jo, Minchul Kim, Hyuchang Choi, and Sukwon Jung

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*liu20893283@khnp.co.kr

Radiological characterization is a crucial component of nuclear power plant decommissioning, as it ensures that contaminated areas are properly surveyed, remediated, and released in compliance with regulatory standards. Traditional radiological surveys are often labor-intensive, time-consuming, and require extensive manual work, particularly in large structures and open land areas. This study presents the design and demonstration of an autonomous system that aims to automate and streamline the radiological characterization process. By integrating an autonomous vehicle with commercially available radiation detection and analysis equipment, the system offers a novel solution to reduce the time and manual effort involved in performing these critical tasks. The autonomous system demonstrated in this study was specifically designed to address the challenges associated with conducting repetitive surveys over large areas. It was tested at the Kewaunee nuclear power plant, where it successfully performed radiological surveys both inside the plant's auxiliary buildings and across expansive outdoor land areas. The system's ability to autonomously navigate these complex environments, coupled with its capacity for radionuclide-specific analysis, proved to be a significant advantage over traditional, manually delivered survey methods. One of the standout features of the system is its capability to identify areas with elevated levels of radioactivity while distinguishing between natural background radiation and contamination from the plant's operations. During the demonstration, the autonomous vehicle was equipped with radiation detectors and analysis systems that allowed it to achieve adequate detection sensitivities for both characterization and final status surveys, depending on the levels of background radiation present in the surveyed areas. The system also demonstrated the ability to autonomously navigate based on GPS data for outdoor surveys and use mapping techniques for indoor environments, enabling it to traverse pre-determined paths without human intervention. This functionality significantly improves the efficiency of the survey process, particularly in large or hazardous areas where manual surveys would be more challenging and time-consuming. The results of the Kewaunee demonstration underscore the potential benefits of automating radiological surveys. By reducing the need for manual labor, the autonomous system decreases worker exposure to potentially hazardous environments while also cutting down on the time and cost required to complete the surveys. In addition, the system's ability to perform radionuclide-specific analysis allows for more accurate characterization of contaminated areas, which is critical for ensuring that all areas meet regulatory site release limits. The automated system can also serve as a valuable tool for monitoring and verifying remediation efforts, providing detailed data that supports decision-making throughout the decommissioning process. In conclusion, the design and demonstration of this autonomous system for radiological characterization mark a significant step forward in the automation of nuclear decommissioning activities. The system's ability to autonomously navigate and perform radionuclide-specific analysis offers substantial benefits in terms of time savings, cost reduction, and improved safety for workers. As the nuclear industry continues to evolve, the adoption of such automated systems is likely to play a key role in enhancing the efficiency and effectiveness of decommissioning projects, ensuring that environmental and safety standards are met while minimizing the challenges associated with manual survey methods. This study provides a strong foundation for future research and development in the field of autonomous radiological surveys, opening the door to further innovations in decommissioning technology.

Keywords: Characterization, Autonomous system, Decommissioning, Final status survey

A Study on Enhancing Clearance of Radioactive Spent Activated Carbon From NPPs Through Parameter Optimization Using RESRAD-Onsite

Seongmin Jo*, Taewoong Lee, Kihyun Cho, Heechul Yoon, Kichul Kim, and Junghyun Lee

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*yuty2104@kps.co.kr

Radioactive spent activated carbon is continuously generated during the operation of nuclear power plants due to periodic replacement. Although most of this waste meets Clearance criteria, a portion marginally exceeds the standards, leading to increasing storage volumes. As the stored quantity increases, it becomes necessary to propose various assessment methods to meet the clearance criteria. In this study, RESRAD Onsite was employed to control various parameters, with the aim of increasing the self-disposal of radioactive spent activated carbon by meeting the required standards. Among the key radionuclides present in the spent activated carbon are C-14, I-129, I-131, and H-3. C-14, with its long half-life and significant impact on internal exposure, was selected as the target nuclide, with the contamination concentration assumed to be 1 Bq/g, which is the self-disposal criterion. Additionally, the most conservative Residential scenario was chosen, and six key sensitive parameters were identified. Controllable variables were then extracted by referring to the main equations for each parameter. To control the density of the unsaturated zone, density increase was assumed by using rollers or similar methods. The results of this evaluation were compared with previous findings. It was concluded that increasing the density of the unsaturated zone affects the distribution coefficient (K_d), density, hydraulic conductivity and total porosity, leading to a reduction in radiation dose and thereby better satisfying the clearance criteria. This research contributes to optimizing the management of radioactive waste from nuclear power plants, potentially increasing the volume of spent activated carbon eligible for clearance through strategic parameter control.

Keywords: RESRAD on-site, DCGL, Spent activated carbon, Clearance level, Economic evaluation

Key Aspects of Demonstrating of Radiation Scanning Robotics in Compliance With the MARSSIM Revisions

Yong-Won Ma*, Se Geun Lee, Dong Hyun Kim, Min Gyu Choi, Yong Guk Son, and Moo-Keun Song

Korea Research Institute of Decommissioning (KRID)

*yongwon@krid.re.kr

With the recent revisions to the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), the inclusion of advanced technologies, particularly mobile robotic systems, has become a pivotal component in modern radiation surveying. The updated guidelines emphasize the importance of these systems for surveying real property, including surface soils and building surfaces, to improve the precision and reliability of radiation surveys, particularly for scoping surveys, characterization surveys, and final status surveys (FSS). Building on the development of robotic systems for nuclear power plant radiation scanning, this paper focuses on the essential steps required to validate and demonstrate these systems, ensuring that testing and verification of the new scanning robots or services utilizing advanced technology can be effectively conducted.

Key elements of demonstration include a thorough examination of the robots' pose characteristic, obstacle detection, obstacle avoidance, and threshold passing performance. While established performance standards and testing methodologies exist for robot locomotion and navigation, it is essential to redefine the testing environments specific to nuclear power plants. This includes adjusting factors such as travel surface conditions, operating parameters, and test procedures.

This paper presents a comprehensive approach to verifying robotic systems for radiation surveying, including controlled laboratory tests, field demonstrations at nuclear decommissioning sites, and real-time automated data collection using GPS, LiDAR, microwave positioning systems, and laser sensors. The goal of this study is to validate the positional accuracy and reliability of radiation scanning robots, ensuring that the position data, coupled with radiation measurements, is consistent and precise. Additionally, radiation detection capabilities must be rigorously tested against traditional sampling methods to ensure that the robots provide data with comparable or superior reliability and resolution. Our aim is not only to enhance the precision and reliability of these measurements but also to provide a viable alternative to traditional radiation surveying methods. This paper demonstrates how advanced robotics can replace legacy technologies by offering superior accuracy, operational efficiency, and adaptability to nuclear power plant environments.

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Keywords: MARSSIM revision, Radiation scanning robots, Robotics validation, Radiation surveying technologies

Tensile Test and Fixture Fabrication for Miniature Tensile Specimens

Wook Sang Jeon*, Young Gwan Jin, Young Jun Kim, and Seung Je Baik

Korea Atomic Energy Research Institute (KAERI)

*wsjeon@kaeri.re.kr

Because of space limitation in most research reactor, specimen miniaturization is necessary. These specimens are tested for mechanical properties and are used for alloy development studies.

Hot cell tensile test of miniature tensile specimens was conducted. This experiment was pre-test for the irradiated specimens. The specimens were machined as SS-3 size for tensile test. It is widely used size for irradiated material tensile testing and is already known to be adequate for reproducibility of tensile data.

Also, all tests were conducted in hot cell, so special tensile fixture was fabricated. The most important function of this fixture is to enable the manipulator to hold the specimen inside the hot cell. It was designed for shoulder gripping so it was able to securely hold the specimen during the test. Tensile specimen was mounted on the fixture by using manipulator. And then it was put on to the rod of tensile machine. So the rod could pull it during the test. And test was conducted at elevated temperature, include room temperature.

You will find that in the stress-strain curve, the slope of the elastic region is often not straight. The actual stress-strain data deviates unlike the ideal case. Therefore, sometimes it is difficult to determine the yield strength after 0.2% offset. Multiple slopes may be considered as modulus of material, and there are several ways for determining modulus. It is possible for referring ASTM E8 example or auto-calculating method of testing machine.

Keywords: Miniature tensile specimen, Hot cell testing, Modulus, Yield strength

Fabrication of Fixture for Specimen Imaging Inside a Hot Cell

Wook Sang Jeon*, Young Gwan Jin, Young Jun Kim, and Seung Je Baik

Korea Atomic Energy Research Institute (KAERI)

*wsjeon@kaeri.re.kr

Imaging the specimen before and after mechanical testing of irradiated specimens inside the hot cell is essential. Before the test, it is necessary to inspect the appearance of the specimens or verify their IDs. After the test, it is common to measure the gauge length of the fractured specimen or analyse the fracture surface. There are several conditions needed to set up digital microscope inside the hot cell. First, all internal electronic components of the digital microscope must be shielded from radiation. Second, the control unit must be located outside the hot cell so that it should be operated outside the hot cell. Digital microscope that meets these conditions is already installed inside the hot cell at KAERI.

Since various types of tensile specimen are tested inside the hot cell, it is crucial to fabricate special fixtures that can take images of them. For these fixtures, several conditions also needed. First, the fixture must be possible to be holded with a manipulator. Since all tests are conducted using a manipulator inside the hot cell, fixture should have a handle that manipulator should be possible to hold and fix it at microscope. Second, the fixture should include a saddle part that the tensile specimens could be holded. So operators could carefully fit the fracture specimens to analyse the fracture or gauge length.

In this experiment, a fixture was fabricated to observe the fracture behavior of small plate type tensile specimens and to measure the length of the fractured specimen. The saddle part was designed considering the width and length of the specimen so that specimen could be easily placed in it. Additionally, to verify the reliability of the length measurement from the digital microscope, a certified calibration block was placed beside the specimen.

Keywords: Fixture, Specimen imaging, Hot cell testing

6분과

방사선환경 및 안전 (ORAL)

Bioaccessibility of Cs- or/Co- contaminated Media Using in Vitro Assay and Human Biokinetic Modeling

Nurul Syiffa Mahzan*, Nur Shahidah Abdul Rashid, Kyungwon Kim, and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*nurulsyiffa@postech.ac.kr, wooyongum@postech.ac.kr

The term “decommissioning” encompasses the administrative and technical actions involved in ceasing the operation and service of nuclear facilities. This comprehensive process entails preparing for dismantling, decontamination, waste disposal, and site restoration. Throughout the decommissioning process, the generation of contaminated concrete (CC) waste occurs, which, despite its low radioactivity, is considered significant due to its substantial volume. Additionally, the concrete may contain cesium (Cs) and cobalt (Co), man-made radionuclides resulting from nuclear activities or accidents. In instances of nuclear power plant decommissioning or severe accidents, inadvertent ingestion of highly dispersible contaminated materials could pose risks to both animals and humans, underscoring the necessity to investigate toxicity and exposure assessment to quantify human intake of Cs and Co through ingestion.

In this study, the bioaccessibility (f_{ba}) of Cs and Co was assessed utilizing the in vitro unified BARGE Method (UBM), where contaminated concrete was spiked with various concentrations of Cs or Co (10-1,000 mg/kg). The acquired data was subsequently used to refine Physiologically Based Pharmacokinetic (PBPK) modeling for predicting bioaccessibility in humans. The bioaccessibility of Cs- f_{ba} in the gastric (GP) and gastrointestinal (GIP) phases ranged from 49.98% to 91.43% and 48.43% to 88.84%, respectively. Conversely, the bioaccessibility of Co- f_{ba} in GP and GIP was between 64.94% to 77.41% and 47.02% to 69.74%, respectively. These outcomes suggest that Cs and Co exhibit greater bioaccessibility in the GP phase compared to the GIP phase, indicating their high solubility in acidic condition. Acidic environments facilitate dissolution reaction that notably promotes the release of Cs and Co from the concrete matrix, particularly involving compounds such as cesium hydroxide (CsOH) and cobalt sulfate heptahydrate ($\text{CoSO}_4 \cdot 7\text{H}_2\text{O}$) or cobalt chloride (CoCl_2), which display heightened solubility in acidic conditions, leading to the dissociation of Cs^+ and Co^{2+} ions into the digestive juices. Following the successful alignment of the model with the Co experimental data, the measured and predicted Co- f_{ba} closely matched for the orally ingesting contaminated concrete samples. To substantiate these findings, a comparison was made between the findings and the results from a literature-based animal study. However, the literature indicated that the predicted value in this study (37%–73%) is higher than in other research results (0.2%–66%), plausibly due to discrepancies in the compound itself. Meanwhile, the biokinetic model for the Cs is still in progress to provide the estimated value of Cs- f_{ba} . As such, an ongoing animal study is being conducted to ensure precision and to provide comprehensive insights.

Acknowledgements

This work was supported by the Korea Hydro & Nuclear Power Co., Ltd (KHNP) Research Fund Haerum Alliance Nuclear Innovation Center.

Keywords: Bioaccessibility, *In vitro*, Cesium, Cobalt, Biokinetic model

Development of Fast Compton/Coded-Aperture Hybrid Gamma-ray Imaging System

Junyoung Lee, Goeun Lee, Taehyeon Eom, Jaeho Cho, Sehyun Jang, and Chan Hyeong Kim*

Hanyang University

*chkim@hanyang.ac.kr

Gamma imaging devices with the capacity to detect and localize radioactive materials can be utilized effectively in a variety of fields, including radwaste management facilities, nuclear facilities, nuclear medicine facilities, and airports and harbors. Despite the development of several portable devices, their small sensor size and low imaging sensitivity severely limit their utilization. To address these limitations, a fast gamma imaging system, the Fast Gamma Imager (FGI), has been developed. This system boasts imaging efficiencies at least several orders of magnitude higher than conventional handheld devices. Furthermore, the FGI demonstrates excellent imaging resolution (less than 10° angular resolution) over a wide energy range (59.6 keV to several MeV). Experiments were conducted to compare the performance of the FGI with that of a Polaris-H420, a commercially available imager suitable for hand-held applications. To compare the performance of the H420, the localization time of the developed FGI was tested at various dose rates (0.001–0.028 $\mu\text{Sv/h}$). The dose rate was controlled by adjusting the position and activity of the ^{137}Cs source. The mean localization time, defined as the average time taken to localize the source from the gamma-ray image over repeated measurements, was evaluated. The number of iterations was set to 100 measurements. The localization time of the H420 was evaluated to be 83 s for a dose rate of 0.028 $\mu\text{Sv/h}$ when a 10 μCi ^{137}Cs source was positioned 1 m in front of the system. Under the same conditions, the localization time of the FGI was only 0.62 seconds, making it 133 times faster than the H420. Even under lower incremental dose rate conditions (0.74 μCi ^{137}Cs source at 2 m or 0.0005 $\mu\text{Sv/h}$ incremental dose rate), the system imaged the location of the source within 300 seconds. In a further field test at the Incheon Airport Customs, the FGI successfully localized a cargo containing a 240 μCi ^{137}Cs source moving along a conveyor belt with 0.15 m/sec speed, showcasing the FGI's exceptional imaging speed and real-time monitoring capability. Throughout these on-site demonstrations, the FGI demonstrated its versatility in various operational settings, offering new perspectives on utilizing radiation imaging for enhanced radiation safety.

Acknowledgements

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Keywords: Gamma-ray imaging system, Compton camera, Coded aperture, High-speed, 3D image

Development of Combined 1D & 2D Numerical Model for Hydrological Transport of Radionuclides

Ugyu Jeong, Kiwon Kang, Youjin Oh, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Potential risks resulting from radionuclides in the environment needs to be evaluated, and its results play an important role in radioactive waste management. Specifically, these evaluations are used for radioactive waste classification, as well as for the risk and performance assessment of disposal sites. To this end, it is necessary to accurately predict the movement of radionuclides through the subsurface over thousands of years. A reasonable transport model should neither underestimate for public health protection nor overestimate for ensuring efficiency and safety of disposal.

In Korea, the RESRAD-ONSITE, a one-dimensional groundwater transport model, is commonly used. However, due to its simplified assumptions, this model may not adequately reflect realistic hydrological behavior. There is a possibility of overestimating radionuclide concentrations, particularly because the advection-reaction based one-dimensional model adopts unreasonable assumptions to account for two-dimensional transport effects.

To overcome these limitations, a combined 1D & 2D numerical model based on the 4th order Runge-Kutta (RK4) method was developed at Ulsan National Institute of Science and Technology (UNIST). This model applies a one-dimensional solver in the unsaturated zone and a two-dimensional solver in the saturated zone, considering advection and dispersion. In addition, it includes a well scenario that accurately evaluates ground water flow and contaminant dilution, considering Cone of depression and Pump & Treat (P&T) models.

In this research, a comparison with the RESRAD-ONSITE and water pathway scenario of the International Atomic Energy Agency (IAEA) was conducted to verify the accuracy of the developed model and to assess concentration tendency according to the improved evaluation methodology. The target nuclide for the comparison is C-14, which has a long half-life of 5,730 years and a non-adsorption property, making it capable of directly being exposed to groundwater and drinking water. Additionally, hydrological conditions were assumed to reflect the domestic environment of clearance, where no engineered barriers are present, unlike in radioactive disposal sites.

As a result, it was confirmed that the concentration of C-14 in the well, calculated by developed model, was relatively lower compared to those of the RESRAD-ONSTE and IAEA models. This result is influenced by the differences in groundwater flow and well modellings, and if the reliability of the developed model is verified, it is expected to contribute to reducing the long-term disposal cost.

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Keywords: Groundwater modeling, Radionuclide transport, Numerical analysis, Radioactive waste management, Subsurface disposal

Development of Compound Tritium Protection Mask

You Ha Lee* and Se Yup Lee

Korea Nuclear Engineering Co., Ltd.

*konecl@naver.com

In heavy-water nuclear power plants such as Wolsong nuclear power plant, tritium accounts for a large amount of radionuclides generated during operation. Tritium has relatively low β -energy, but it is harmful because tritium usually exists in the form of tritiated water vapor in air and it is absorbed into the body at the same speed as general water when inhaled into the body. Therefore, portable protective equipment is required to minimize the radiation exposure of workers from tritium and other radionuclides.

A compound tritium protection mask has been developed to improve the efficiency of work, the convenience of using the mask, and the field work conditions and work efficiency by supplementation for the various shortcomings of the tritium protection mask currently used in heavy water power plants. It consists of a face mask, a breathing hose, an ice container, a connector, and a filter. The face mask has improved face adhesion and respiratory resistance, and the breathing hose is a lightweight material that makes it easy to decontaminate and dry, and the ice container is a lightweight material that makes it easy for workers to move, lasts more than two hours to cool, and prevents backflow of water generated in the breathing tube. In addition, the connector is a one-touch quenching type that is easy to detach each part, and the filter is detachable that can remove radioactive particles in usual field working conditions.

Prototype compound tritium protection mask was manufactured through a design and production of molds for each component, a partial modification of the molds, injection molding processing and assembling of the components produced.

As a result of the performance test on tritium removal for the prototype compound tritium protection mask, it was estimated that the tritium removal efficiency of the compound tritium protection mask was maintained at more than 90% until 90 minutes have elapsed, indicating that the tritium removal efficiency is satisfactory enough to protect the workers. In addition, even after 90 minutes, about 60% of the ice remains in the ice container and the tritium removal efficiency was maintained at more than 90%, it means that it can be worn for at least 90 minutes.

It is expected that the compound tritium protection mask could contribute to the enhancement of the safety of workers in the field.

Acknowledgements

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Keywords: Tritium, Tritiated water vapor, Tritium protection mask, Tritium protection respirator

Dose Conversion Coefficients for Radiation Workers in Various Exposure Scenarios

Min Chae Kim^{1,*}, Sora Kim¹, Yoomi Choi¹, Jiyeon Kim¹, Byung Il Min¹, Kyungsuk Suh¹, and Chan Hyeong Kim²

¹Korea Atomic Energy Research Institute (KAERI)

²Hanyang University

*minchaekim@kaeri.re.kr

This study investigates the impact of exposure scenarios on dose conversion coefficients (DCCs) by utilizing posture-modified computational phantoms and different exposure geometries. Accurate radiation dose assessment is critical in both routine dosimetry and accident scenarios, particularly when various postures are involved. Traditional radiation protection often assumes a standing posture for dose estimation; however, this can lead to inaccuracies in scenarios where workers or victims adopt postures like kneeling, squatting, or bending. To address this gap, different working postures were developed using the Mesh-type Reference Computational Phantom (MRCP) published in the International Commission on Radiological Protection (ICRP) 145 report.

Simulations were conducted using the Geant4 code, focusing on calculating absorbed doses for whole-body (D_{WB}) and personal dosimeter (D_{Det}). The study explored exposure geometries such as anterior-posterior (AP), posterior-anterior (PA), left and right lateral (LLAT and RLAT), rotational (ROT), isotropic (ISO), and more complex scenarios like bottom-to-top (BT) and top-to-bottom (TB) contamination, which simulate ground and ceiling contamination, respectively. Different radioisotopes, including Ir-192, Cs-137, and Co-60, which are common sources in radiological accidents, were used for the exposures.

Results showed that posture has a significant effect on doses and DCCs. In AP exposures, body bending reduced the D_{WB} by up to 31% compared to a standing posture, while PA exposures showed even more pronounced effects, with DCCs reaching as high as 2.3 in medium-bent postures. Lateral exposures demonstrated a significant reduction in D_{Det} due to arm and thigh shielding, with DCCs varying from 0.5 to 1.7 across postures. For BT and TB contamination scenarios, squatting and kneeling postures were found to have a major influence on dose distribution due to body compression and shielding effects. The findings emphasize the importance of considering postures in dosimetry, particularly in complex radiation fields, to improve risk assessment and safety in radiation protection.

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Keywords: Dose conversion coefficients, Ground and ceiling contamination, Monte-carlo simulation, Mesh-type reference computational phantoms, Posture-modified computational phantoms

Estimation of Residual Fission Product Radioactivity at North Korea's Punggye-ri First and Sixth Nuclear Test Sites

Hongmo Park

Korea Institute of Nuclear Safety (KINS)

hongmopark@gmail.com

North Korea conducted six nuclear tests at the Punggye-ri Nuclear Test Site from the first nuclear test in October 2006 to September 2017. The first nuclear test, which was an atomic bomb, had the smallest nuclear explosion yield, while the sixth nuclear test, which was a hydrogen bomb, had the highest nuclear explosion yield. In establishing a radioactive waste management policy related to the North Korean nuclear test site, assessing the radioactivity present in the nuclear test site is an essential element.

In this study, the radioactivity of the remaining fission products in the first and sixth nuclear test sites was estimated. The radioactivity of the fission products which generated as a result of the nuclear test is directly related to the nuclear explosion yield. The nuclear explosion yield of the underground nuclear test is estimated by a unique algorithm from the seismic detection agency. In this study, the highest estimated nuclear explosion yield was selected from the estimated nuclear explosion yields of various agencies, and the radioactivity was calculated by estimating 2 kilotons for the 1st nuclear test and 280 kilotons for the 6th nuclear test.

The other crucial factor is the fission yield by fast neutrons of Pu-239 and U-235 for the atomic bomb and the fission yield of U-238 14 MeV neutron for the hydrogen bomb. The nuclear explosion yield could convert into the number of nuclear fissions which converted into the number of atoms by the fission yield. The number of atoms of each decay chain's nuclides was calculated over time for atomic numbers 83-155 and converted into radioactivity. In the case of the first nuclear test, it is presumed that the bomb used Pu-239 as fissile material, and the radioactivity of the fission products remaining 18 years after the nuclear test was calculated alongside U-235. In the case of the sixth nuclear test, the radioactivity of the fission products remaining 7 years after the nuclear test was calculated by assuming an 80% contribution of U-238 fissile material and a 20% contribution of Pu-239 or U-235.

In the case of the first nuclear test site, ten nuclides exceeded 1.0 Giga Becquerel and four nuclides including Sr-90 remained above 1.0 Tera Becquerel. In the case of the sixth nuclear test site, twenty nuclides exceeded 1.0 Giga Becquerel radioactivity, and three nuclides including Nb-93m remained above 1.0 Peta Becquerel.

Keywords: North Korea nuclear test site's radioactivity, Nuclear weapon test, Fission yield, Nuclear explosion yield

Research on the Development of a Mobile Facility for Radioactive Laundry Waste Treatment Using Surfactants Concentration Technology and Photooxidation Technology

Dong-Yeon Kim, Sung-Hyeon Kim, and Do-Hyun Kim*

NILEPLANT.Co., LTD.

*nileplant2024@gmail.com

Radioactive Laundry waste generated from nuclear power plants contains surfactants, which capture radioactive nuclides and form polymer micelles. Conventional radioactive ion capture technology cannot completely remove radionuclides contained in laundry wastewater, so they are diluted and discharged as is when the liquid effluent discharge standards are met. However, since surfactant concentration and decomposition technologies have been developed to remove radioactive nuclides from radioactive laundry wastewater, it can be seen that discharging radioactive laundry wastewater as it is is not appropriate from an ALARA perspective.

A typical surfactant is an organic substance that has a chain-like hydrophobicity structure composed of carbon and hydrogen, and a hydrophilic functional group at the end. Recently UV oxidation technology using photocatalyst has been developed and used for various purposes as a technology to decompose organic matter, and is also being utilized as a technology to decompose surfactants. Since the concentration of surfactant components contained in radioactive laundry wastewater is only about 100 ppm, in order to effectively decompose the surfactant components contained in radioactive laundry wastewater by applying UV oxidation technology using a photocatalyst, it is necessary to concentrate the surfactant concentration to 2,000 ppm or more and process it.

Therefore the experimental mobile radioactive laundry wastewater treatment device manufactured through this study is largely composed of a surfactant concentration module using membrane separation technology, a surfactant decomposition module based on photocatalytic O_3/H_2O_2 UV oxidation technology, and a cationic radionuclide removal module that removes radioactive particle components and cationic radionuclides from the radioactive laundry wastewater from which the surfactant components have been removed.

In the surfactant concentration module, the surfactant component is concentrated and transferred to the surfactant decomposition module, and the permeate is transferred to the radioactive nuclide removal module to remove radioactive cation nuclides and then discharged. The surfactant concentrated waste liquid transferred from the surfactant concentration module has the surfactant component removed in the surfactant decomposition module to a level that does not affect the operation of the ion exchange resin. And the concentrated waste liquid from which the surfactant component has been removed and the permeate that has passed through the surfactant concentration module are transported to the radioactive nuclide removal module to remove radioactive particle components and radioactive cation components and discharge them.

Currently, a performance test has been conducted using a mobile radioactive waste treatment device for testing, and the device will be improved based on the performance test results to produce a commercial mobile radioactive waste treatment device.

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Keywords: Treatment of radioactive waste, Decomposition of surfactants, Concentration of surfactants, Photocatalyst reaction

6분과

방사선환경 및 안전 (POSTER)

Design Guideline and Performance Evaluation Methods for Radiation Monitors With Small Size and Light Weight Power Supply

Sung-Ho Lee

Korea Atomic Energy Research Institute (KAERI)

leesungho@kaeri.re.kr

A power supply in a radiation monitor is responsible for converting ac power from utility grid into low-voltage dc power. In general, the power supply is divided into a linear power supply and a switching mode power supply (SMPS) according to the power conversion method. The linear power supply has the advantage of very low output voltage fluctuation and low electromagnetic interference noise, but has also the disadvantage of being bulky and heavy; this might increase potential radioactive wastes. On the other hand, the SMPS has the quite smaller size and lighter-weight. This paper provides detailed guidelines for the SMPS design. The circuit topology of SMPS is based on flyback, which is widely used in low power applications such as radiation monitors. This paper also explains in detail method to evaluate the performance of SMPS for radiation monitors designed and built based on technical specifications such as IEC-61000 and IEC-60532. Finally, the 30 W prototype for the SMPS of the radiation monitor is built and tested for the performance evaluation.

Acknowledgements

This work was supported by Korea Atomic Energy Research Institute (KAERI) R&D Program (No. 522430-24).

Keywords: Low radioactive waste, Power supply, Switching mode power supply, IEC-61000, IEC-60532, Radiation monitor

Development of the First Principle Model for Fault Detection and Diagnosis on the Heating, Ventilation, and Air-Conditioning (HVAC) System in the Nuclear Fuel Cycle Facility

Hyojik Lee^{1,*}, Wonseok Oh², Woo Jin Jo¹, Sang Jin Park¹, and Seung Nam Yu¹

¹Korea Atomic Energy Research Institute (KAERI)

²Hanbat National University

*hyojik@kaeri.re.kr

The Korea Atomic Energy Research Institute (KAERI) operates various nuclear fuel cycle facilities. Among the safety-related structures, systems, and components (SSCs) of these nuclear fuel cycle facilities, those critical to confining high-level radioactive materials include the hot cells and the Heating, Ventilation, and Air-Conditioning (HVAC) system, which form the confinement boundaries of the hot cells. A failure in the elements of the HVAC system could compromise the negative pressure function of the hot cells, potentially leading to a radioactive material leakage incident. Consequently, the department operating the facilities conducts periodic inspections of the critical components of the HVAC system to maintain the negative pressure in the hot cells. However, since these inspections rely heavily on the judgment of the operators, there is a lack of evidence to support maintenance decisions, and inconsistencies may arise depending on the individual operators. As a result, it is challenging to perform maintenance based on reliable evidence through monitoring and diagnosing the condition of the HVAC system. Currently, most of the facility inspection records are manually recorded, resulting in very limited data available to assess operational conditions. Therefore, the types and amounts of data recorded are insufficient to perform Fault Detection and Diagnosis (FDD) based on data-driven condition monitoring.

To address these shortcomings, the development of a physics-based model grounded in first-principles modeling is being pursued. By utilizing this physics-based model, it is possible to monitor the operational status of the SSCs and apply it to fault detection and diagnosis. The development of the physics model is being carried out for the Irradiated Materials Examination Facility (IMEF), including the HVAC systems, and involves whole-building energy modeling to simulate how well the HVAC systems control the temperature and pressure in each zone of the building. The physical modeling uses EnergyPlus, developed by the DOE, to perform dynamic analysis of heat conduction, radiation, and convective heat transfer based on thermal and material balance. To achieve this, geometrical modeling of the IMEF has been performed, dividing zones according to configuration, material, function, and heat load. Although EnergyPlus allows for HVAC system modeling, this physical model development utilizes Dymola, based on Modelica, which provides a library for HVAC components. The chiller and boiler, which supply heat to the facility, are modeled using the chilled water supply temperature and steam temperature and flow rate provided by the institute's utilities as boundary conditions. Currently, the connection between EnergyPlus and Dymola is being established, and the developed model will undergo technical verification, followed by validation through parameter adjustments of the physical model to match the measured operational data, ensuring it closely resembles actual operating conditions. Once validated, the physics-based model will provide insights into various fault scenarios and help address the issue of insufficient fault operation data.

Acknowledgements

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Keywords: First principle model, Physical model, EnergyPlus, Dymola, Fault Detection and Diagnosis (FDD), Whole building energy model, Heating Ventilation and Air Conditioning (HVAC)

Effect of Irradiation on Micro Hardness of Baffle Former Bolt Removed From Pressurized Water Reactor

JinHo Park^{1,*}, Young Gwan Jin¹, Seong Sik Hwang¹, and Seungjin Oh²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*jh@kaeri.re.kr

To improve the safety of nuclear power plants, research on the effect of high neutron irradiation on the mechanical properties of metallic materials is very important. Baffle former bolts are crucial for maintaining the structural integrity of the reactor core, and it is vital to understand how irradiation impacts their mechanical properties.

This study examines the effect of irradiation on the micro hardness of a baffle former bolt removed from a pressurized water reactor, Kori unit 1. Micro hardness testing was conducted on different sections of a single bolt that experienced varying levels of irradiation. The results reveal variations in micro hardness depending on the irradiation exposure, demonstrating the material's varying responses to different radiation levels.

These findings will contribute to a deeper understanding of the behavior of reactor components under irradiation and are expected to aid in the assessment and enhancement of reactor safety and performance.

Keywords: Baffle former bolt, Irradiation, Micro hardness, Pressurized water reactor

Development of an Autonomous Mobile Robot for Detecting Harmful Gases and Radiation in Confined Spaces

YongTae Shin^{1,*}, JongGyu Kim¹, and WonSeok Kim²

¹CROWN Technol. Hold. Co., Ltd.

²Pohang University of Science and Technology (POSTECH)

*service4u@hanmail.net

Nuclear power plants have many containment vessels, and these containment vessels may emit water vapor, hydrogen, carbon monoxide, and other harmful gases, so monitoring of these gases is necessary, and oxygen monitoring is necessary to prevent suffocation in confined spaces. Additionally, since radiation may occur, radiation monitoring is necessary.

When operating or decommissioning a nuclear power plant, the level of radiation contamination of used equipment or generated waste is manually determined, exposing workers to risks. Accordingly, there is an urgent need for rapid radiation measurement in radiation-contaminated areas through the introduction of a real-time telemetry robot (unmanned system) to nuclear power plants and contaminated areas.

In this study, a self-driving robot was developed to measure harmful gases and radiation in confined spaces. The main research results are as follows.

- Design and manufacture of a remotely operating robot: Gas sensors (H₂S/O₂/CO₂/CO), radiation (γ-ray) sensors, and autonomous driving-related sensors (IMU sensor, RPLiDAR, camera, wired/wireless router, antenna, ultrasonic sensor, etc.) were installed with the consideration of mobility of the robot.
- Portable control HMI: An HMI that can monitor harmful gases (H₂S/O₂/CO₂/CO) and radiation measurement values and robot movement path was manufactured.
- Development of robot operating algorithm: Autonomous driving was realized by applying ROS (Robot Operation System) noetic-based G_mapping SLAM (Simultaneous Localization And Mapping) and Navigation.
- Development of signal processing algorithm: Data acquired through the signal processing algorithm included radiation measurement values, harmful gas measurement values, remote robot running speed and movement path, and LiDAR measurement values.
- Selection of a specific confined space: The Radiochemistry Laboratory of the Division of Advanced Nuclear Engineering at POSTECH, which is an accessible confined space, was selected. Robot operation, radiation measurement, and harmful gas measurement tests were performed.

In addition, in cooperation with the radiation environment team of the Korea Radioactive Waste Agency, an on-site test was conducted on a low- and medium-level radioactive waste storage facility (August 2, 2024), and during a radioactive disaster prevention drill scheduled for November 2024, radiation is planned to be measured at the radioactive waste storage facility.

Acknowledgements

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Keywords: Radiation, Harmful gases, Telemetry robot, Autonomous driving, ROS

Preliminary Field-Test of Main Component Diagnosis System of HVAC Equipment in Fuel Cycle Facility

Sang-Jin Park*, Eui Kwang Jeong, and Seungnam Yu

Korea Atomic Energy Research Institute (KAERI)

*sangjin@kaeri.re.kr

HVAC (Heating, Ventilating, and Air-Conditioning) systems are ubiquitous in modern buildings and industrial plants. These systems are often considered minor components in facility operations, as their malfunctions typically result in temporary discomfort for workers or equipment rather than causing catastrophic damage to the facility. However, in certain fuel cycle facilities, particularly those containing hot-cells, the HVAC systems play a crucial role and they must maintain negative pressure conditions in hot-cell rooms continuously throughout the year, serving as a vital part of the containment system for radioactive materials. Consequently, preserving the integrity of HVAC equipment in fuel cycle facilities is of paramount importance.

This study presents a preliminary field-test of a main component diagnosis system for HVAC equipment in a fuel cycle facility. The focus is on the fan system, a key component of HVAC equipment, which consists of an impeller, electric motor, belt, and pulley. We employ a vibration-based diagnosis system to inspect the fan system. Utilizing this diagnostic approach and adhering to ISO 20816-3 standards, we conduct a preliminary examination of the characteristics and integrity of the operational fan system within the actual working field of the fuel cycle facility.

Acknowledgements

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Keywords: Fuel cycle facility, HVAC system, Vibration-based diagnosis, ISO 20816-3

Introduce of the Performance Procedures of the Whole Body Counter

Si Young Kim*, YongHo Jin, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*siyoung.kim@khnp.co.kr

To monitor the internal exposure due to the radionuclide remaining in the worker's body, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) operates Whole Body Counters (WBCs) according to the plant procedures. KHNP conducts the quality control activities for the WBC following the plant's procedures. As a part of quality control activities, KHNP conducts an inter-comparison test of the performance of WBC between nuclear power plants site once a year. The inter-comparison test is performed according to American National Standards Institute (ANSI) N13.30 report published in 2011. The ANSI N13.30 recommends or requires test categories, radionuclide for each test, radioactive concentration of radionuclide, phantom for each test and performance standard. KHNP conducts total body test among the test categories and uses four radionuclides including ^{54}Mn , ^{60}Co , ^{134}Cs , ^{137}Cs . And the radioactive concentration of each radionuclide is greater than 3 000 Bq and the difference of concentration between radionuclide is less than three times. The RMC- II (Radiation Management Corporation) phantom is used instead of BOMAB (Bottle Manikin Absorber) phantom for total body test. The performance standard is Root Mean Squared Error (RMSE) and criteria is 0.25 or less.

Keywords: Whole body counter, Performance test, ANSI N13.30, Nuclear power plant

Optimization of the Scintillator Geometry in a Phoswich Detector for Alpha Activity Assessment in Nuclear Emergency

Jaehun Lee*, Hyeonmin Lee, and Hee Reyoung Kim

Ulsan National Institute of Science and Technology (UNIST)

*jhlee21@unist.ac.kr

This study focuses on optimizing the geometry of a scintillator for alpha detection in a Phoswich detector based on MCNP simulation. ^{241}Am with a half-life of 432 years, which is one of the strong alpha emitter generated from nuclear power plant accidents providing a suitable alpha radiation source in the laboratory for extended periods, was chosen as a target nuclide for alpha detection. Using the composition and density data of ^{241}Am , a simulation was set up in MCNP.

To determine the optimal geometry, MCNP simulations were compared with analytical solutions derived from existing formulas. In the process, the energy of the alpha particles emitted from ^{241}Am was determined using a weighted average value of 5.371 MeV (with 12.8% at 5.443 MeV and 85.2% at 5.486 MeV). Using this value, the range of alpha particles in a ZnS (Ag) was calculated to be 0.00306 cm.

The analytical calculation was performed before constructing the detector and alpha particles was indicated to be sufficiently detected in the ZnS (Ag) cell. However, too much thick ZnS (Ag) cell could make beta radiation also lose its energy in that cell, which resulted in beta radiation not clearly being detected in the rear cell of ZnS (Ag), potentially reducing the reliability of detecting radiation beyond this point. Therefore, the MCNP tool was used to optimize the detection thickness.

The simulation process was carried out for a structure where a thin layer of ZnS (Ag) was placed on the front surface, followed by a layer of BC404 for beta detection and a layer of GAGG (Ce) for gamma detection, aligned linearly in sequence behind the alpha detector. Considering the existing detector structure, ZnS (Ag) was modeled in a cylindrical shape, and its thickness was optimized through simulation. The optimal thickness of the ZnS (Ag) layer was determined to be 8 μm , allowing for the complete detection of alpha particles while minimizing alpha detection in the BC404 cell along with little absorption of beta and gamma particles in the ZnS (Ag) cell.

The analysis showed the relative difference of 73.86% between the analytical calculation and simulation, which was attributed to increased uncertainty caused by simplifications in the integration process used to calculate the range of alpha particles. These findings indicated that the present study on optimizing the scintillator geometry in a Phoswich detector would provide a basis for the confirmatory design of the alpha detection system with enhanced efficiency.

Acknowledgements

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Keywords: MCNP, Phoswich detector, Nuclear emergency, Alpha scintillator

Precautions for Using the Internationally Applied Offsite Dose Calculation Code for Nuclear Power Plants (NRCDose3) - Suspected Errors and Causes

In Gyu Chang and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

The radiation dose for the members of the public near nuclear power plants (NPPs) in the Republic of Korea is calculated according to the regulatory guideline 2.2 of the Korea Institute of Nuclear Safety (KINS), which is based on the United States Nuclear Regulatory Commission's (U.S.NRC) Regulatory Guide 1.109, Rev.1. KINS developed the Integrated Dose Assessment Code package (INDAC) based on this guideline, while operators have developed and utilized the K-Dose 60 code. However, since both codes are proprietary and not publicly accessible, the NRCDose3 code (GASPAR, LADTAP, and XOQDOQ), which is also based on Regulatory Guide 1.109 and aligns with KINS guideline 2.2, is frequently used in radiation dose assessment research.

NRCDose3, developed by the U.S. NRC's Radiation Protection Computer Code Analysis and Maintenance Program (RAMP), is used in the licensing process of NPPs in the U.S. and was also employed in the Design Control Document (DCD) of the Republic of Korea's APR1400. Given that the U.S. radiation protection system is based on International Commission on Radiological Protection (ICRP) 2, earlier versions of NRCDose3 code included only one dose coefficient. However, subsequent updates have allowed users to select dose coefficients based on ICRP-30 or ICRP-72.

NRCDose3 has undergone continuous updates, with Version 1.1.4 released in March 2024, and the previously draft NRCDose3 Manual (NUREG-2267) was published in July 2024. In this study, we evaluated radiation doses using dose coefficients as ICRP-72, which align with the Republic of Korea's radiation protection system (based on ICRP-60), and performed validation through manual calculations using Microsoft Excel®.

During this process, five errors were identified in NRCDose3 code Version 1.1.4, including errors in dose coefficients, ingestion and inhalation rates, and the failure to display certain output results (four in GASPAR and one in LADTAP). These errors resulted in incorrect radiation dose calculations. This study analyzes these errors and their causes, stressing that users of NRCDose3 code Version 1.1.4 and earlier versions should be aware of these issues and exercise caution in their assessments until the code is updated.

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Keywords: Radioactive effluent, NRCDose3, Safety assessment, Errors, Root cause analysis

A Study on U-235 Enrichment Measurement Technique Without Suitable Standard Materials: Challenges, Analysis, and Alternative Methods

Woojin Kim* and Jinho Ryu

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*kimwj@kinac.re.kr

This study explores advanced methodologies for determining the enrichment levels of Uranium-235 in contexts where appropriate standard reference materials are unavailable. Accurate measurement of nuclear material enrichment is essential for ensuring adherence to safeguards agreement and domestic regulatory frameworks. Nonetheless, the absence of standard materials presents substantial challenges to achieving accurate and reliable measurement results.

The research begins by analyzing the limitations of an existing method under non-standard conditions, identifying key areas where accuracy is compromised. To address these issues, the study proposes alternative measurement techniques that can be used in the absence of standard materials. These techniques involve the analysis of the gamma peaks duplicated in narrow energy bands and the use of detector characterization and computed efficiencies. Through experimental validations, the effectiveness of these alternative methods is demonstrated. These efforts are aimed at enhancing the reliability of U-235 enrichment measurements and ensuring consistency across different operational contexts. Furthermore, the study provides a comprehensive evaluation of the practical applicability of the alternative methods. It discusses the potential advantages and limitations of these methods in real-world applications, offering insights into how nuclear material verification can be improved through more accurate enrichment measurement techniques.

The findings of this research hold considerable significance for the fields of nuclear safeguards verification, as they lay the groundwork for enhanced measurement practices in scenarios in the absence of standard materials. The study concludes by proposing future research directions aimed at refining these methodologies and resolving any residual challenges, thereby contributing to improved regulatory compliance and the safer management of nuclear materials.

Keywords: Safeguards, U-235 enrichment measurement, Standard materials, Peak analysis

Development of Safe Handling Facilities and Technologies for Tritium Metering and Dispensing

Yong-mok Cho, Seung-chul Shin, and Soon-hwan Son*

AGENCORE CO., LTD.

*shson@agencore.net

This work aims to develop advanced facilities and technologies for the safe metering and handling of tritium gas. To reinforce safety and reliability in handling radioactive isotopes, the work has applied design principles of deep defense, diversity, and independence. The tritium handling facilities have received radioactive isotope sales permit from the Nuclear Safety and Security Commission (NSSC) and can handle up to 400,000 Ci of tritium annually. Tritium will be supplied from the tritium removal facility in Wolsong nuclear power plant of KHNP. The tritium handling facilities consists of four main subsystems: tritium dispensing part, tritium monitoring part, tritium removal part, and exhaust/drainage part. The tritium dispensing part includes a heater, metering tanks (1,000 Ci and 5,000 Ci), a cryogenic Gas Chromatography for hydrogen isotope analysis, and a vacuum pump. The tritium monitoring part utilizes a Tritium Monitor to continuously monitor tritium contamination levels and exhaust gases within the glove box, with a measurement range of 0 to 199,999 $\mu\text{Ci}/\text{m}^3$. The tritium removal part consists of both the metal getters and catalytic recombiner. The exhaust/drainage part includes a 10 HP, 8,700 CMH blower and three 1,000 L capacity retention tanks. Under normal operation, the facilities minimize tritium emissions following the ALARA (As Low As Reasonably Achievable) principle and aims to prevent radioactive leaks even in abnormal situations. The deep defense design against tritium leaks comprises five stages: a metal getter bed containing depleted uranium, the glove box, a tritium removal system (TRS) for removing contaminated air inside the glove box, and a real-time monitored exhaust/discharge system. The glove box meets ISO-10648-2 (Containment Enclosures) Class 1 or higher standards with a leakage rate of less than $10^{-4}/\text{h}$. The technologies developed in this work effectively prevent the leakage of radioactive materials during the tritium handling. It is expected that these technologies will significantly enhance the safety of tritium handling in the future. The next step is to obtain a sealed radioactive isotope production license from the Nuclear Safety and Security Commission using these facilities and technologies.

Acknowledgements

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Keywords: Tritium, Glove box, Tritium removal system, Tritium metering and dispensing

Dose Rate Estimation Based on Radioactivity Analysis Scenario in Hot Cell Facility

SeongHee Park*, SeungUk Yoo, and MoonOh Kim

KONES Corp.

*shpark@kones21.com

The Korea Research Institute of Decommissioning (KRID) was established to enhance domestic nuclear decommissioning technologies and improve the quality of radiological analysis. The facility, currently under construction, includes hot cell facilities designed for the physical and chemical characterization of intermediate level waste (ILW) generated during nuclear power plant decommissioning.

The design and operational methods of similar domestic and international facilities have been referenced in the development of this facility. However, since this hot cell is specifically designed for the analysis of intermediate-level waste, its purpose differs from that of existing facilities, making it difficult to directly apply operating method of oversea to domestic hot cell.

Accordingly, operating methods suitable for domestic hot cell facilities and samples to be handled are continuously being established. These operating methods should ensure the radiation safety of workers, but after the initial design, there is a lack of radiation safety assessments for workers considering various types and levels of samples according to the operating methods and work scenarios.

In this study, we evaluated the radiation dose distribution within the analysis facility for both normal and abnormal scenarios involving intermediate-level waste analysis, while also calculating unit doses based on the positions of samples and workers and developing a database to enable rapid preliminary dose assessments by estimating expected work routes and times.

These data are expected to serve as a resource for minimizing worker radiation exposure during the analysis of intermediate-level waste.

Acknowledgements

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Keywords: Decommissioning, Intermediate level waste, Hot cell, Dose rate estimation

Development of a TDCR System Using Digital Coincidence Counting

Tanapol Dachviriyakij^{1,2,*}, Jong Man Lee², Byoung Chul Kim², Kyoung Beom Lee², Yeong Jae Park²,
Dong Hye Heo², Tae Hee Kim², and Ka-Young Park²

¹University of Science and Technology (UST)

²Korea Research Institute of Standards and Science (KRISS)

*tanapol.d@kriss.re.kr

The TDCR (Triple to Double Coincidence Ratio) technique is a well-established method in liquid scintillation counting (LSC) that allows for the absolute determination of beta-emitting radionuclides without the need for an external standard. Beta events were detected by three of photomultiplier tubes (PMTs) positioned in 120 degrees to each other. The method involves measuring the triple and double coincidence rates, coincidence counts are determined and then the ratio of triple to the logical sum of double coincidence counts which are used to infer the detection efficiency. In this study, a specific type of LSC cocktail, Ultima Gold (UG), was employed for its compatibility with the beta decay process and its effectiveness in maximizing scintillation light output. The beta-emitting isotopes of interest in this work include Fe-55 which are low-energy beta emitters. Fe-55 decays via electron capture to Mn-55. It was chosen because it is one of key isotopes associated with the management of low and intermediate level radioactive waste.

The experimental setup included the use of Hamamatsu R331-05 photomultiplier tubes, which are high quantum efficiency and fast response times for low-energy beta emitters. The signals from these PMTs were digitized using the NKFADC 125-Mini, a model of Flash Analog-to-Digital Converter (FADC) designed for high-speed data acquisition. This FADC model features a sampling rate suitable for capturing the fast, transient signals generated during the scintillation process. Preliminary results indicated that the FADC system can effectively give the beta counts as same as the conventional system. In conclusion, the integration of FADC in the TDCR method for measuring beta-emitting isotopes Fe-55 has proven to be an effective approach. This methodology improves radioactivity measurements in various applications, including environmental monitoring, nuclear safety, and medical diagnostics. The future plan involves integrating the developed data acquisition method with NaI (TI) scintillation counters to establish a coincidence counting system.

Keywords: Triple to Double Coincidence Ratio (TDCR), Flash Analog-to-Digital Converter (FADC), Low-energy beta emitters

Biokinetic Modeling of U Based on ICRP Occupational Intakes of Radionuclides

Hyun Woo Lee^{1,2}, Si wan Noh², Cheol Hyun Kang², Seung Kyu Lee², In Su Chang², and Yoon Sun Chung^{1,*}

¹Hanyang University

²Korea Atomic Energy Research Institute (KAERI)

*ychung@hanyang.ac.kr

The primary pathways of uranium absorption are inhalation in workplaces and ingestion in daily life. As well as uranium is a radioactive nuclide that undergoes alpha decay, uranium is classified as a naturally occurring radioactive material (NORM). As the enrichment of ²³⁵U increases, mass ratio of ²³⁴U with a higher specific activity also rises. Therefore facilities engaged in the handling of highly enriched uranium, such as those involved in the processing of nuclear fuel, must consider both the chemical toxicity and the internal dose resulting from internal exposure due to uranium intake.

The assessment of internal exposure consists of two principal processes. The first step is to estimate the amount of intake, and the second is to calculate the dose coefficient [Sv/Bq], which is the proportionality factor between intake and dose. Biokinetics is indispensable for both the estimation of intake and the calculation of dose coefficients. In comparison to relying solely on literature references, in-house biokinetic model can be implemented to various potential scenarios that may occur in the actual workplace, thereby enhancing the reliability of dose assessment and achieving more proper radiation protection results.

In this study, we developed a biokinetic model for uranium with reference to ICRP Publications 2, 69, 78, 100, 130, 137, and the research paper of A. Birchall and A. C. James (1989). Specifically, we obtained data on compartment models using first-order kinetics of transfer rates between compartments from the ICRP documents and employed the methodology presented by A. Birchall and A. C. James (1989) to solve the system of ordinary differential equations using exponential matrix, thereby enabling us to calculate the retention of uranium in each compartment over time. The model was validated by comparing the retention of ²³⁸U over time in both inhalation and ingestion cases with the results calculated using the OIR Data Viewer, which is provided as an appendix in the ICRP OIR Series. The maximum relative error was below 3% in all cases.

After validation process, we added the capacity to differentiate by age and gender, as well as to simulate acute and chronic intake. Subsequently, for uranium with the same absorption type, the residual amount in each compartment was compared based on the type of intake, age group, and gender. In all cases, uranium primarily accumulated in the skeleton, liver, and kidneys. In all cases, uranium primarily accumulated in the skeleton, liver, and kidneys. For ingestion intake, the differences between 3-month-old infants and other age groups were more pronounced compared to inhalation intake. This is presumed to be primarily due to the fact that the gastrointestinal absorption fraction of uranium in 3-month-old infants is 4%, which is double that of other age groups. Additionally, among the radioactive substances deposited in the respiratory system, the fraction actually absorbed within the respiratory tract was only up to 16.8% of the total. The remaining portion was either expelled externally through particle transport or transferred to the digestive system. As the absorption rate decreased, the fraction absorbed within the respiratory system further diminished.

The model developed in this study can be applied to routine and special monitoring of workplaces and workers who handles uranium. Additionally, model suggested in this study can be applied to the calculation of uranium retention and excretion in individuals who chronically consume uranium through drinking water and food. Furthermore, the methodology is broadly applicable to other radionuclides, allowing the development of models that can be modified for various parameters, such as gender, age group, AMAD/AMTD, and absorption type.

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Keywords: Uranium, Internal exposure, Biokinetics, Age-specific model

Safety Analysis of Radiological Exposure Due to Melting Furnace Explosions in Wolsong Nuclear Power Plant's Radioactive Waste Facility

Sim-Gwon Wee* and Chang-Lak Kim

Kepco International Nuclear Graduate School (KINGS)

*kazanjs@naver.com

The decommissioning process of a nuclear power plant involves the melting of metals using a furnace. During this process, external factors such as humidity can trigger furnace explosions, leading to the inevitable dispersion of radioactive materials through gas flow. Meteorological data is crucial for assessing the movement and dispersion of radioactive materials released into the environment during a nuclear accident. Airflow data is essential to understanding how much radiation people might be exposed to. Therefore, it is necessary to study what happens when radioactive materials are released into the air during the shutdown of a nuclear power plant.

This study focuses on determining the potential radiation exposure levels in the event of an explosion at the Wolsong Nuclear Power Plant. To predict how radiation would disperse in the first two hours following a hypothetical accident, meteorological data from 2020 to 2023 and a specialized tool called PAVAN, developed by the U.S. Nuclear Regulatory Commission (U.S. NRC), were used. The guidelines for gas dispersion scenarios are legislated by the Nuclear Safety and Security Commission (NSSC) and are based on U.S. NRC standards. This research analyzes radiation dispersion using meteorological data from the Wolsong site.

The results of the radiation dispersion simulation conducted with the PAVAN program provide insights into how parameters such as wind direction, speed, and atmospheric stability impact the area surrounding the incident site. The study's results indicate that radiation levels at all Wolsong reactors do not exceed the safety limit of 1 mSv. This suggests that the radiation risk to nearby workers is minimized.

The decommissioning of Wolsong Unit 1, the world's first pressurized heavy water reactor (PHWR) to be decommissioned, is a significant global event. This project represents a major advancement in the field of nuclear decommissioning. The insights and understanding gained through this process will be highly beneficial for future PHWR decommissioning projects. Successfully completing this project will position Korea as a leader in nuclear waste management and serve as a strong example for other nations.

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Keywords: Atmospheric dispersion, Wolsong nuclear power plant, Nuclear decommissioning, Melting furnace explosion, Radiological exposure, Pressurized Heavy Water Reactor (PHWR)

EPR Culture Survey of Emergency Personnel at Nuclear Research Facilities

Sohyeon Lee*, Bongseok Kim, and Hyun Ki Kim

Korea Atomic Energy Research Institute (KAERI)

*sohyeonlee@kaeri.re.kr

Since the International Atomic Energy Agency (IAEA) released the EPR-Method in 2003, it has been working on a draft revision as of 2024, with plans to publish the updated version in 2025. The draft emphasizes the importance of ‘Emergency Preparedness and Response (EPR) Culture’. EPR Culture is defined as “an assembly of competence, beliefs, commitments, behaviors, accountability and appreciation of EPR roles and responsibilities resulting in EPR receiving the attention warranted by its significance”. IAEA highlights that conducting training, drills and exercises supports the EPR Culture. Accordingly, this study investigated the impact of emergency preparedness training on EPR Culture for employees engaged in radiological emergency organization at KAERI. The education was conducted for a total of 266 emergency response personnel from April 1 to June 14, 2024, and the satisfaction survey was administered at the end of each training session. The survey included 10 questions designed to assess EPR Culture, with responses evaluated using a 5-point Likert scale. Analysis of the responses revealed a significant correlation between radiological emergency response training and the promotion of EPR culture. Experienced senior emergency personnel demonstrated a high level of interest in radioactive disaster prevention and actively participated in the training. Meanwhile, junior emergency personnel with less experience showed relatively higher risk perception scores for nuclear research facilities. In conclusion, it is essential to enhance the practical curriculum to increase participation in training and ensure accurate recognition of facility risks. These results are expected to contribute to the quality management of radiological emergency response education and training programs at KAERI.

Acknowledgements

This study has been conducted as a part of the project of emergency preparedness for nuclear research facilities in KAERI.

Keywords: Nuclear or radiological emergency, Emergency preparedness and response, EPR culture

Derivation of Dose Rate Correction Factor of Mobile Detector According to Scanning Rate

Hyeonmin Lee*, Seung Hun Shin, and Hee Reyoung Kim

Ulsan National Institute of Science and Technology (UNIST)

*leehm@unist.ac.kr

The speed correction function for the radiation dose rate of a mobile detector, which moves along with a vehicle, was derived based on simulations and experimental characterization. In the event of the radiological disaster caused by a nuclear power plant accident, radiation exposure to the public can occur due to the release of radioactive nuclides into the atmosphere. The nuclides released during the early stages of an accident diffuse into the air. Residents living near the nuclear power plant evacuate using vehicles according to emergency evacuation plans. During the evacuation process, external dose management for the residents is necessary, and immediate response is required to identify risks due to radioactive contamination along the evacuation routes. This study covers the derivation of speed-dependent correction factor for moving detectors, where the decrease in counting rate due to vehicle speed causes the dose rate reduction. It eventually aims for the development of the mobile detector that can be mounted on evacuation vehicles to measure the dose rate more accurately by compensating for the decrease in dose rate during moving of the vehicle.

First of all, dose rate was measured experimentally for the moving point source and compared with simulation, showing that the relative difference was less than 2.5% for all evaluated speeds (0 to 10 km/hr). An MCNP simulation was used to derive a speed correction function for the actual accident environment with radiation cloud based on the moving point source characterization. Detection efficiency was statically derived at various distances, and fitted and averaged over time to obtain the compensation matched to dynamic status.

The accident site simulation modeled the source term in a hemispherical shape, focusing on key radioactive nuclides such as Cs, I, Xe, and Kr. The distance between the ground and the detector was set to 100 cm, and the source term sizes were set at 30, 60, and 100 meters. Speed correction functions derived from these simulations are expected to be applied to actual dosimeters, enabling simultaneous evacuation and measurement for more accurate dose rate assessments.

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Keywords: MCNP, Dosimetry, Mobile detector, Scanning rate correction factor, Nuclear disaster

Comparative Review of Overseas Off-Site Dose Calculation Code Status

Seok-Ju Hwang*, Si-Young Kim, Hang-Rae Cho, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hsjhhhhh@khnp.co.kr

Domestic nuclear power plants conduct Radiological impact on the Environment Review (RER) every year in accordance with the nuclear safety and security commission (NSSC) Notice No. 2017-17 and disclose the results to the public. In order to evaluate the impact on residents of the contents of RER, for domestic nuclear power plants, evaluations are conducted using K-DOSE60 (developed by KEPRI, 2012). Currently, KHNP is developing various packages for overseas nuclear power plants exports. In the field of radiation environment, a comparative review of the status of dose calculation code for overseas exports is also required.

In the United States, evaluations are conducted using NRCDOSE3 and RESRAD developed by nuclear related organization such as NRC, DOE and ANL. These codes can evaluate not only impact on human but also impact on biota. In addition, unlike in Korea, most nuclear facilities are located inland, so lake, river and estuary can be added to the exposure pathway due to liquid effluents to conduct evaluations.

In Europe, many countries are developing dose calculation codes and using them for evaluation. In France, EDF, CEA and Andra use SYMBIOSE, CERES, ERICA and ARIA IMPACT, Andra MOM codes. Spain uses the CROM8 code developed by CIEMAT. Slovakia uses ESTE code developed by ABmerit, and UK uses PC-CREAM developed by RAD to conduct resident dose calculation. Among them, ERICA codes cannot be used for human impact evaluation, but are used for biota evaluation. In Europe, it is also possible to assess exposure pathways for lake, river and estuary for liquid effluent evaluation because it is located inland.

It is expected to be used in research on adding a liquid effluent evaluation pathway for future overseas exports and on biota impact evaluation methods.

Keywords: RER, ODC (Off-site Dose Calculation), Liquid effluent

Analysis of Individual-Based Dose Assessment Methods for Public Living Near Nuclear Power Plant

Byung Min Lee*, Jae Seok Kim, and Min Seok Park

Korea Institute of Radiological and Medical Sciences (KIRAMS)

*byungmin95@kirams.re.kr

The dose assessment for public living in the vicinity of nuclear power plants (NPPs) is typically conducted on a source-based approach, as opposed to an individual-based approach. This is because the radionuclides and the amount of their release from the NPPs is known, and in addition, various exposure pathways can be considered. However, in some countries, individual-based dose assessment methods are being employed for various research purposes, including the long-term health impact on individuals. Those methods enable the investigation of the relationship between radiation dose and health effects in public residing near NPPs. The objective of this study was to analyze the methods for individual-based dose assessments that could be applied to epidemiological studies in Korea.

In Ontario, Canada, as part of a study analyzing the health impacts for public living near NPP, effective dose was analyzed using tooth-electron paramagnetic resonance (EPR) dosimetry. The tooth-EPR dosimetry is a method that evaluates the dose contained in tooth enamel using EPR dosimeter. In the study, a total of 64 tooth samples were collected and analyzed for doses. The evaluation determined that the annual effective dose rate for public from artificial radiation sources (such as NPP effect and medical exposure) was 0.63 mSv/y.

In Fukushima, Japan, personal dosimeters were distributed to public to evaluate the dose following the Fukushima Daiichi nuclear accident. The personal dosimeter used was 'D-shuttle' equipped with global positioning system (GPS) functionality. The evaluation determined that the annual dose rate for public in the Fukushima area was mostly below 1 mSv/y, with no significant values measured.

Both cases provided the dose assessment methods and dose results for the public, but the dose results were not significant enough to investigate the relationship between radiation dose and public health effects. Moreover, their dose assessments focused on the limited exposure pathways, not considering the internal exposure. The further study should be performed to investigate the relationship between the radiation dose and public health effects. For that, these findings may serve as foundational data for studies analyzing the long-term dose assessments on the public living near NPP.

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Keywords: Dose assessment, Public, Nuclear power plant, Individual-based

Comparison of the Impact for Dose Assessment According to Use the Latest Statistic Data

Nayun Choi¹, Soyun Kim¹, and Jae Hak Cheong^{2,*}

¹Korea Institute of Nuclear Safety (KINS)

²Kyung Hee University

*jhcheong@khu.ac.kr

Liquid and gaseous effluent discharged from nuclear power facilities causes radiation exposure to residents, and it is limited by the dose criteria. In Republic of Korea, the Korea Institute of Nuclear Safety (KINS) presents the methodology and default values for dose assessment through Regulatory Guide 2.2 (Exposure Dose Assessment for Residents), but the usage factor, which reflects the lifestyle of the residents, has not been revised since KINS/GR-199 (2000), so it is difficult to say that it considers the current lifestyle characteristics in 2024.

A study to revise the usage factor was conducted through KINS/RR-808 (2011), but it is not known whether it has been updated or reflected. In addition, the 'Discharge Plans of Liquid and Gaseous Radioactive Materials from Nuclear Power Plant and Related Facilities' approved this year states that the input factors will be reviewed every five years and re-evaluated if necessary to ensure the quality of the discharge limits - derived based on the dose assessment results.

In this study, the latest data published by Statistics Korea (2022) used to compare the internal exposure dose by ³H and ¹⁴C, which have a high contribution to dose by effluent discharged from domestic nuclear power plant. Compared to the intake value in Regulatory Guide 2.2, the intake of grains, meat, etc. increased by up to about 2 times, and it can be expected that it will have a significant change in domestic exposure dose.

Since the usage factor is a value that directly affects the exposure dose, it may become necessary to periodically investigate, review, and reflect the lifestyles of residents similar to discharge quantities and site characteristics data for more realistic dose assessments in the future.

Keywords: Radioactive effluents, Dose assessment, Statistic, Usage factor

Performance Test Results of Reading Capability for Personal Dosimeters of Nuclear Power Plants Last Five Years

Si Young Kim*, YongHo Jin, and Jung-Kwon Son

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*siyoung.kim@khnp.co.kr

To monitor individual exposure of Nuclear Power Plants (NPPs) workers, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) operates personal dosimeter reading systems for each plant site. All of domestic NPPs use Thermo-luminescent dosimeters (TLDs) for personal dosimeter. According to Article 115, Paragraph 1 of the Nuclear Safety Act Enforcement Decree, dosimeter reader facilities must conduct performance test by the regulatory body. KHNP conduct inter-comparison test for TLD reader capability between NPPs every year, before the regular performance test of the regulatory body. For the inter-comparison test, the performance indicators and criteria such as a bias (≤ 0.35), standard deviation (≤ 0.35), and acceptable limit (≤ 0.50) are reflected American National Standards Institute N13.11 requirements which published in 1983. And also, among the eight test categories, seven test categories were performed excluding neutron test category. In the last five years (2019~2023) all NPPs have been confirmed to meet performance criteria for seven test categories. Through these results, it was confirmed that the quality of the external exposure dose evaluation results of NPP workers was properly maintained.

Keywords: Personal dosimeter, Nuclear power plant, External exposure, Performance test

Development of a VR Training Program for Validation of Internal Contamination Monitoring Standard Operating Procedure (SOP)

Kihoon Kim and MinSeok Park*

Korea Institute of Radiological and Medical Sciences (KIRAMS)

*chulipak@kirams.re.kr

In a radiation emergency, internal contamination may occur due to radiological release into the environment. In order to prevent the spread of internal contamination, it is necessary to implement the radiation protection measures. The monitoring program for affected population is an efficient method for radiation emergency preparedness, it should be prepared based on the standard operating procedure to ensure the accuracy and effectiveness of radiation emergency preparedness. For that, we developed several standard operating procedures for measuring the internal contamination of individuals. In the case of radiation emergency preparedness, however, it should consider the field-applicability of standard operating procedure because practical measurement of internal contamination in the initial phase of accident might be performed in the field. Therefore, the present study aims to develop the VR-based training program for verifying the standard operating procedures we developed and improving the effectiveness of education and training for radiation responders. To verify the internal contamination monitoring standard operating procedure, a VR training program was developed that allows implementation of internal contamination monitoring procedure on actual situation.

The program implemented the RIIDEye (NaI detector) from Thermo Scientific, which is distributed to radiation emergency medical institutions nationwide for internal contamination monitoring. The program was composed of training courses divided into equipment installation, equipment operation, and actual operation. The equipment installation training described in detail the processes of connecting the detector and the analysis device, assembling the shield, and connecting the PC. The equipment operation training consisted of training on operating the analysis device and the PC program, respectively. Each course was divided into stages of equipment calibration - background measurement - sample measurement - spectrum analysis, and detailed procedures for performing were inserted for each stage. In addition, it included not only simple equipment operation procedures for measurement, but also detailed additional explanations for each screen and procedure. The equipment operation training was distinguished as thyroid measurement training and urine sample measurement training, so that training for each measurement situation could be intensively trained. In addition, a practical course was prepared to distinguish internal contamination based on the measurement results, so that it could be familiar in an actual emergency monitoring situation.

We believe that the developed training program can improve the effectiveness of education/training for radiation responders. Moreover, the standard operation procedures can be applied considering the practical measurement conditions in the field. The developed VR training program will be applied to actual education/training, and improvements to the standard operating procedure for internal contamination monitoring will be derived and reflected based on the feedback.

Acknowledgements

This study was supported by a grant from the Korea Institute of Radiological and Medical Sciences (KIRAMS), funded by the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 2104038).

Keywords: Internal contamination, Radiation emergency, Monitoring, Standard operating procedure

Addressing the Detrimental Impact of Cooling System Malfunction and Long-Term Inactivity on Quantulus 1220 Background Levels

Myeong Hoon Joo^{1,2,*}, Sang Hyun Jeong^{1,2}, Young Ju Song¹, Doo Cheol Kim¹, Ji Yang So¹, Seong Joo Kang¹,
Sang Wook Lee¹, Sung Hyun Joo², Jae Gyeong Choi², and Byung In Min²

¹DSENTEC Inc.

²In-je University

*myeonghoon.joo@dsentec.com

The Quantulus 1220 liquid scintillation counter (LSC) is recognized for its ability to measure low background counts per minute (cpm) due to the use of thick lead shielding, allowing for more accurate and faster radionuclide analysis compared to the HIDEEX system. The background cpm of the Quantulus 1220 is approximately 1-3 cpm, significantly lower than the HIDEEX SL-300, which measures 13-16 cpm for 3H in full spectrum. However, there are factors that can increase the background cpm in this instrument. A significant portion of the background signal in LSCs originates from cosmic rays, environmental gamma radiation, and residual radioactive impurities in the detector materials, leading to crosstalk due to phenomena such as photomultiplier tube (PMT) fluorescence and Cerenkov radiation.

The Quantulus 1220 instrument currently in use was manufactured in 2011 and had been stored in a sealed environment for 5-6 years without use, accumulating dust and suffering from a malfunctioning cooling system due to prolonged inactivity. In such a condition, the buildup of alpha- and beta-emitting progeny from radon gas decay within the lead shield can increase the instrument's background cpm during operation. Additionally, the malfunctioning cooling system has led to increased temperatures in the sample tray, preventing adequate sample cooling and resulting in thermal noise and chemical reactions within the samples, further contributing to elevated background noise. These issues have led to the 3H full-spectrum background cpm of the Quantulus 1220 rising to approximately 5-7 cpm. The prolonged period of inactivity has also resulted in channel shifting events, which have detrimentally impacted the measurement capabilities and detection efficiency of the instrument.

If samples are analyzed under these conditions, with a minimum detectable activity (MDA) value set at 5 Bq/L, the measurement time could increase from approximately 300 minutes to over 600 minutes, doubling the required time for analysis. This study addressed these challenges by monitoring the background cpm of the instrument and conducting internal calibration using NIST Standard Reference Material (SRM 4927 g). As a KOLAS-accredited radioactivity analysis laboratory, tritium calibration was performed following the procedures outlined by the Korea Association of Standards & Testing Organizations (KASTO) in the document "Standard Calibration Procedure of Liquid Scintillation Counters" (KASTO 18-80213-367). The process involved the preparation and use of quenching sources for internal calibration. The final calibration certificate was issued, and the results were compared with previous calibration data to analyze changes in efficiency calibration.

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Keywords: Liquid Scintillation Counter(LSC), Tritium (3H), Background, KOLAS

Feasibility Assessment of Two Dosimeter Algorithm Based on Machine Learning

Soo Hyeok Lee¹, Hwi Joon Jeong¹, Kyungmin Kim¹, Hyungtaek Kim², and Kyung Taek Lim^{1,*}

¹Sejong University

²Korea Atomic Energy Research Institute (KAERI)

*kl2548@sejong.ac.kr

A personal dosimeter is designed to reflect the amount of irradiation received by workers in the field. However, such dosimeters can underestimate the actual radiation exposure as typical workers wear them on their chest only. In particular, the dose equivalent indicated by the dosimeter may underestimate or overestimate the worker's effective dose in inhomogeneous radiation fields, which cannot be measured directly. This issue is notably critical in environments like steam generator (S/G) water chambers, where high-dose, inhomogeneous radiation fields are common during nuclear power plant overhauls. Overestimation of the personal dose equivalent can lead to extended overhaul periods and an increase in the number of workers, as work time is managed to ensure that dose limits are not exceeded.

The Two Dosimeter Algorithm (TDA) is often employed to compensate the personal dose equivalent more accurately in inhomogeneous radiation fields. In this study, we modeled the working environment of an S/G water chamber based on MCNP6. In particular, a MIRD Phantom was positioned in the chamber, equipped with personal dosimeters on the worker's chest and back. By varying the position and orientation of the MIRD Phantom, we acquired 16 sets of personal dose equivalent values for the chest and back and the corresponding effective doses for the MIRD Phantom. The standard uncertainty was calculated using the relative error of the MCNP results, and the data was subsequently augmented for machine learning training. The performance of the machine learning-based Two Dosimeter Algorithm (TDA), developed using data derived from MCNP simulations, was evaluated by comparing it with Thermoluminescent Dosimeter (TLD) readings from workers in the S/G water chamber at the Yonggwang Nuclear Power Plant.

We evaluated the feasibility of using machine learning to rapidly and accurately determine the correlation between effective doses received by workers and the measurements from front and back dosimeters. This approach is expected to make a significant contribution to advancements in radiation safety management.

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Keywords: Machine learning, Two dosimeter algorithm, Thermal luminescence dosimeter, Radiation protection, MCNP

Development of Visualization Algorithms for Monitoring National Radiation Dose Rate Changes in the Event of a Nuclear Accident

Jae Wook Kim¹, Min Beom Heo¹, Hee Kwon Ku¹, Hyun Jin Boo², Tae Hee Kim², Byung Gi Park²,
and Sang Hun Shin^{1,*}

¹FNC Technology Co., Ltd.

²Soonchunhyang University

*ssh9431@fnctech.com

The impact of nuclear accidents is not confined to local areas; it can spread widely depending on wind and weather conditions. Therefore, it is crucial to monitor and respond to nationwide changes in radiation dose rate in the event of an accident. Efficient processing and visualization of radiation data are essential for supporting rapid decision-making during such incidents. This study has developed a visualization algorithm that enables the observation of changes in the national environmental radiation dose rates over time in the event of a nuclear accident in South Korea or neighboring countries, such as China.

The visualization algorithm was developed using Python, utilizing radiation dose rate data provided by the Korea Institute of Nuclear Safety (KINS). The data are structured in a DataFrame format, with rows representing dates or times and columns indicating radiation dose rates by region. Visualization was performed using the Pydeck library. The radiation dose rates are displayed on a map using a color gradient ranging from green (indicating low levels) to red (indicating high levels), allowing for intuitive assessment of risk levels. Additionally, a user interface was created, enabling users to set desired time periods and observe continuous visualizations of spatial and temporal changes in radiation dose rates.

Currently, the algorithm visualizes the radiation dose rates as discrete points, which limits visibility in areas where monitoring posts are not installed. To address this limitation, future studies will explore the application of interpolation methods or the implementation of heatmap formats. Additionally, by incorporating actual measured airborne radiation data by altitude, the algorithm is expected to be applicable for early prediction in the event of a nuclear accident in neighboring countries.

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Keywords: Visualization algorithm, Radiological emergency preparedness, Python program, Environmental radiation dose rate

Design and Assessment of Aerosol Sampling System According to ANSI N13.1 for Nuclear Facilities

Sung-Ho Lee

Korea Atomic Energy Research Institute (KAERI)

leesungho@kaeri.re.kr

According to NWB 3000 and ANSI/ANS 55.4, airborne radioactive substances released into the environment from stacks and ducts of nuclear facilities such as nuclear power plants shall be continuously monitored by a device, and the monitoring results shall be recorded. In order for effluent radiation monitoring, radiation monitors are commonly used to monitor radioactive particulate, iodine, and noble gas by continuously collecting samples of gases emitted through the stack or duct in nuclear facilities. In order to collect representative sample, a sampling system for the stack or duct monitor consisting of a sampling nozzle and a sample transfer pipe is required. ANSI N13.1 presents an approach to representative sampling. The standard provides guidelines and performance criteria for the proper use of air sampling nozzles, transport lines, sample collection and monitoring devices, and gas flow measuring methods in obtaining valid measurements.

This paper introduces the criteria presented in ANSI N13.1, the detail design method about a nuclear facility in Korea Atomic Energy Research Institute (KAERI), and the practical application for the stack effluent monitor.

Acknowledgements

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Keywords: Aerosol sampling system, Nuclear facilities, ANSI N13.1, Shrouded nozzle, Deposition

Development of a Marine Environment Exposure Dose Assessment Model for Marine Radiological Accident

Sora Kim, Dong-Kwon Keum, Yoomi Choi, Jiyeon Kim, Minchae Kim, Hyeon-Jeong Kim, Byung-Il Min,

Kihyun Park, and Kyung-suk Suh*

Korea Atomic Energy Research Institute (KAERI)

*kssuh@kaeri.ac.kr

There have been several cases of marine contamination due to the direct release of radionuclides from nuclear facilities into the sea or the deposition of radionuclides in the air to the sea surface. In particular, after the discharge of contaminated water from the Fukushima nuclear power plant, there is a great concern about radiological effects on people due to marine radioactive contamination.

A dose assessment model has been developed to evaluate the radiological doses to the public following marine radiological accidents. The dose assessment code considers six exposures pathways, as follows: external exposure through beach activity, swimming, and boating and internal exposure through ingestion of seawater during swimming, ingestion of seafood, and inhalation via sea spray. Three age groups were considered: infants, children, and adults. About 60 radionuclides, including Cs, Sr, Pu, and H3, were considered.

Average radionuclide concentrations for the sea surface layer and all depths of seawater in each marine compartment are selectively applied as source-term data, depending on the exposure pathways. Radionuclide concentrations in seafood (three marine organism groups: fish, invertebrate, seaweed) are calculated based on the equilibrium model. By assuming a representative individual who receive high radiation exposure, conservative values were applied to the life characteristics and dietary habits of the representative individual. Concentration factors were taken from IAEA/TRS-422 (2004). External dose coefficients were obtained from Federal Guidance Report No. 15 (2019). Ingestion and inhalation dose coefficients were obtained from International Commission on Radiological Protection (ICRP) Publication.

As a next step, it is planning to build a module that can predict the radiological doses to the public by linking the dose assessment model with the marine dispersion model, and apply it to the case study for evaluating the public exposure following a hypothetical marine radiological accident around the Korean peninsula.

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Keywords: Marine radiological accident, Marine radioactive contamination, Marine environment dose assessment model, External exposure dose, Internal exposure dose

A Study on the Thorium in Urine Analysis by ICP-MS Following Nitric Acid Dilution

Cheol Hyun Kang¹, Hyun Woo Lee^{1,2}, Jong Il Lee¹, and In Su Chang¹

¹Korea Atomic Energy Research Institute (KAERI)

²Hanyang University

*chkang1@kaeri.re.kr

Thorium (Th) is a naturally occurring radioactive element which can be found at trace levels in soil, rocks, water, plants and animals. The radiation workers, especially who are dealing with thorium isotopes, can intake material through inhalation. ²³²Th can be used for individual monitoring for thorium isotopes, because it is the majority of thorium isotope. The urine analysis for in-vitro analysis and the lungs analysis for in-vivo measurements shall be used for the ²³²Th compounds individual monitoring. However, for lung measurement, ²²⁸Ac is measured to assess ²³²Th. The detection limit is relatively higher than that of alpha spectrometry or Inductively Coupled Plasma-Mass Spectrometry (ICP-MS).

American National Standard and Health Physics Society (ANSI/HPS) recommends the validation of in-vitro radiochemical procedure to assure the reliable data produced by the laboratory from the “Performance Criteria for Radiobioassay (ANSI/HPS N13.30-2011)”. There are several procedures that can validate the analysis of uranium in human excreta, such as bias, precision, MDA, average recoveries and suggested categories by the Multi-Agency Radiological Laboratory Analytical Protocols Manual (EPA 2004).

The Urine Analysis Laboratory in the Korea Atomic Energy Research Institute (KAERI) devised the validation procedure of analyzing ²³²Th in urine with the nitric acid dilution and the ICP-MS through this study. The surrogate urine samples with cautiously spiked an instrument calibration standard which contains trace amount of thorium. The concentration of 33.9, 89.3, 210.8, 506.6, 1014.2 ng/L has been used for the validation procedure. The results were compared and evaluated with the criteria of ANSI/HPS N13.30-2011.

Thorium was measured by NexION 2000B (PerkinElmer, USA) quadrupole ICP-MS system. Because there is Total Dissolved Solid (TDS) around 7,000 ppm in urine, it cannot be measured by itself without any dilution procedure or the pre-treatment procedures to minimize the matrix effect of the urine samples. The dilution procedure is relatively simple way of measurement, compare to the acid digestion or the resin separation. In this study, 10-fold nitric acid dilution has been used.

The samples and the calibration solutions were diluted with 2% (v/v) nitric acid. A calibration curve at the concentration of 5-10-20-50-100 ng/L was created prior to the sample measurements. To remove the memory effect, 0.025 mol/L in 5% (v/v) nitric acid was used as the rinse solution. Thorium was determined at the mass number of 232 using ²⁰⁹Bi as an internal standard.

The validation results were within the range suggested in the ANSI/HPS N13.30-2011. With the developed method, this laboratory can analyze thorium handling worker's urine routinely and occasionally. For further work, the ICP-MS setting for optimal measurement will be proceeded to get the reliable data from human urine.

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Keywords: In-vitro bioassay, Urine, Thorium, Concentration measurement, ICP-MS, Validation

Enhancing Environmental Monitoring: LTL System Development in PRIDE

Youngwoo Kwon*, YoungSang Lee, and Seonho Noh

Korea Atomic Energy Research Institute (KAERI)

*kyw@kaeri.re.kr

PRIDE is composed of an integrated system linking the following: engineering-scale oxidation pretreatment, electrolytic reduction, electrolytic refining, and recycling and immobilization of salt waste. PRIDE cell equipments support operations of pyroprocess equipments in the argon cell and supply required utilities to the cell. The argon cell is on the second floor in PRIDE. The argon cell measures $40.3\text{ m} \times 4.8\text{ m} \times 6.4\text{ m}$. The argon cell is on the second floor in PRIDE. The argon cell measures $40.3\text{ m} \times 4.8\text{ m} \times 6.4\text{ m}$.

PRIDE cell equipments include large and small equipment transfer systems, two gravity tubes, in-cell crane, trolley hoist etc. All cell equipments were designed to be remotely operated and maintained.

The utility system of PRIDE was developed for controlling the concentration of impurities (oxygen $< 100\text{ ppm}$, moisture $< 100\text{ ppm}$) in the argon cell.

LTL (Large Equipment Transfer system) is to transfer large and small equipment to/from argon atmosphere cell. Capacity of large equipment transfer system : $2.2\text{ m (D)} \times 2.2\text{ m (H)}$, 2.8 ton. It has purging system for exchanging gas interlocked control.

To maintain the Ar-cell atmosphere, an oxygen moisture analysis system was developed to analyze the impact on the LTL and in-cell environment. This paper shows how the system is configured and can be used in Ar-cell and LTL.

Review of Graded Approach for Environmental Monitoring and Assessment Activities in Nuclear Emergency

Bongseok Kim* and Hyun Ki Kim

Korea Atomic Energy Research Institute (KAERI)

*kbs@kaeri.re.kr

According to IAEA GSR Part 7, nuclear facilities are categorized into 1 to 5 categories depending on the type of facilities and accident scenarios. This method of categorization is based on assessing the hazard level of the nuclear facility and potential accident scenarios. Consequently, the scope of an accident's impact is dictated by its category. Thus, the required emergency response activities are tailored according to the category of the nuclear facilities. Specifically, in the field of environmental monitoring and evaluation, emergency response activities need to be adapted according to the type of accident and the extent of its impact. Therefore, it is necessary to review emergency response activities in environmental monitoring and evaluation relative to the nuclear facilities category.

However, the current radiation emergency response organization does not provide the necessary activities that are subdivided according to the category of nuclear facilities. Even though various categories of nuclear facilities coexist at places, the existing emergency response organization fails to reflect current situation. Consequently, emergency response organizations in the field of environmental monitoring and assessment should be reconsidered of the methods that can be applied to different categories of nuclear facilities.

The IAEA's EPR-Method suggests necessary organizations for environmental monitoring and assessment, including the Aerial Survey Team (AST), Radiation Monitoring Team (RMT), Radionuclide Identification Team (RIT), Assessment and Advisory Team (AAT), Emergency Sampling Team (EST), and Laboratory Facility (LAB). It also recommends a differential application by categorizing the necessary organizational responses by country and facility category.

A graded approach aligned with nuclear facility categorization is also essential for emergency response organizations, and it is crucial to have adequate human resources, considering the functions and scope of activities of these organizations. Particularly, given the framework of Korea's nuclear emergency response and preparedness, where national-level environmental monitoring and evaluation activities are initiated upon declaration of a site area emergency, such considerations for emergency response organizations should be examined at the national level. Therefore, forming an emergency response organization based on response functions and providing training on these functions is required.

Acknowledgements

This study has been conducted as a part of the project of emergency preparedness for nuclear research facilities in KAERI.

Keywords: Nuclear emergency, Radiation, Environmental monitoring, Graded approach

Study on Applicability of Smear Paper Through Machine Learning for Beta Nuclide Identification

Min Ji Kim* and Hee Reyoung Kim

Ulsan National Institute of Science and Technology (UNIST)

*minji4644@unist.ac.kr

The study explores the identification of beta nuclides adapting smear papers by using machine learning models. The development of systems for detecting and monitoring beta radiation is crucial for ensuring the safety of workers in nuclear facilities, particularly due to the risks associated with internal exposure to radiation. Building on a pre-feasibility assessment conducted with disk sources, this study further applies the methodology to smear data. This primary focus was on enhancing the identification of beta nuclides by analyzing spectrum data obtained from smear papers. Machine learning algorithms were employed to accurately identify different beta nuclides based on their energy spectra, which were analyzed to determine the detection probability of each nuclide.

The experiment utilized a CAM 450 model, which is a high-resolution passively-implanted planar silicon (PIPS) detector. The nuclides used were ^{60}Co , ^{90}Sr , ^{137}Cs , and ^{152}Eu . The selection of nuclides is based on their significance in environmental radioactivity assessment and research: $^{90}\text{Sr}/^{90}\text{Y}$ are prominent high-energy beta emitters, ^{60}Co and ^{137}Cs are key gamma emitters with additional beta emission, and ^{152}Eu is important as a fission product and for use in standards and biochemical research. After evaluating each individual nuclide, assessments were also performed for 15 different combinations of multiple nuclides. The nuclide was identified based on the input spectrum data using supervised classification learning. Among the supervised classification methods, the machine learning models used in this study included random forest, gradient boosting, and support vector machine models. Spectrum data were collected over 15-minute interval, resulting in a total of 200 data points, which were split into training and test datasets in an 80:20 ratio.

The results from the machine learning models showed a 100% identification probability for both support vector machine and random forest models, and 99.5% for the gradient boosting model. With the successful validation of smear data applicability, this approach could be effectively used in laboratory settings for cases of unidentified contamination. The process would involve wiping the contaminated area with smear paper, then inputting the data into the machine learning models to accurately identify the nuclide.

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Keywords: Beta nuclide, Identification, Machine learning, Random forest, Gradient boosting, Support vector machine

Monte Carlo Simulations to Investigate the Coincidence Summing Correction Factors for Urine Bioassay

Kihoon Kim, MinSeok Park*, and Minsub Kim

Korea Institute of Radiological and Medical Sciences (KIRAMS)

*chulipak@kirams.re.kr

A urine bioassay is an in-vitro method used to assess internal exposure to radioactive materials, offering high sensitivity for detecting low levels of radioactivity in samples. Gamma spectrometric analysis is commonly applied in urine bioassays to measure gamma-emitting radionuclides, as it provides a non-destructive and efficient analysis method. However, the close geometry of volumetric sources in gamma spectrometry can lead to the issue of coincidence summing effects (K_5), where two or more gamma photons are detected simultaneously, resulting in an erroneous summation of energies. This phenomenon can significantly impact the accuracy of activity measurements. Therefore, accounting for coincidence summing is crucial to enhance the precision of activity estimation. The objective of this study is to determine the coincidence summing correction factors for a coaxial high-purity germanium (HPGe) detector.

The HPGe detector (GC3018, Canberra Industries, Meriden, CT, USA) used at the Korea Institute of Radiological and Medical Sciences was modeled using the Monte Carlo N-Particle (MCNP) transport code developed by Los Alamos National Laboratory. Counting efficiencies were calculated using a tally function in conjunction with the full width at half maximum (FWHM) of the detector. The study considered cylindrical samples (U8, 90 ml) and 1 L Marinelli beaker samples. Validation of the simulation geometry was performed by comparing the counting efficiencies obtained from both experimental measurements and simulations.

To calculate the correction factors, the peak-to-total (P/T) ratio was derived as a function of gamma energies. Monte Carlo simulations were employed to compute the total efficiency for specific gamma-ray energies. Based on the simulation results, coincidence summing correction factors were determined. These factors were then applied to correct the activities of radionuclides (including ^{134}Cs and ^{60}Co) prone to coincidence summing. Furthermore, the correction factors were validated through participation in domestic and international intercomparison programs for gamma spectrometric analysis.

Implementing coincidence summing correction in gamma-ray spectrometry is essential for ensuring accurate quantitative analysis. The results of this study demonstrate that the calculated correction factors can significantly improve the accuracy of activity estimation in urine bioassays, reducing uncertainties induced by the coincidence summing effect.

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Keywords: Urine bioassay, Monte carlo simulation, Coincidence summing effect, Gamma spectrometry, High-purity Germanium

Performance Evaluation of Wireless Communication for Drone-based Aerial Radiation Monitoring System

Min Beom Heo, Jae Wook Kim, Hee Kwon Ku, and Sang Hun Shin*

FNC Technology Co., Ltd.

*ssh9431@fnctech.com

Since the Fukushima nuclear accident in 2011, the necessity of aerial radiation measurement has been highlighted, demonstrating that it is an essential response for national safety, public health, and environmental protection. Recently, the development of the unmanned aerial vehicle (UAV) industry has enabled various research efforts to measure aerial radiation at significantly low costs. In particular, the development of wireless communication technology is essential for real-time data collection. During the development of wireless communication, various factors such as drone payload, power consumption, data volume, and transmission distance must be considered. Performance evaluation is also essential in scenarios involving long-distance measurements or measurements in areas that are difficult for humans to access, because the likelihood of data loss increases in these conditions.

In this study, a multi-channel radiation detector was developed to measure real-time aerial radiation. The detector is equipped with four sensors positioned in all directions (east, west, south, and north) to track the source of the radiation. The data throughput measured by each sensor is large, as it includes energy spectrum information. The data transmission distance for measuring aerial radiation ranges from a few meters to several hundred meters. To meet these requirements, a wireless communication system was developed using two technologies, LTE and Zigbee, allowing for selective use depending on the situation. Additionally, data packets were optimized to minimize data loss and to enable real-time data transmission.

Performance evaluations were conducted to ensure data reliability and to assess data loss rates. Multiple variables, such as the presence of interference, measurement distance, and the number of measurement sensors, were considered to collect data and compare it with wired data. The performance evaluation results showed that LTE provided stable communication performance with little influence from interference and distance. In contrast, Zigbee experienced some data loss depending on interference and distance.

Therefore, in long-distance operations such as real nuclear accidents, it is expected that primarily using LTE and utilizing Zigbee as a supplementary communication method in environments without interference can help minimize data loss. This dual communication strategy offers both reliability and flexibility, ensuring that critical data is transmitted effectively even under challenging conditions. Additionally, by equipping devices with storage options like USB for auxiliary data storage, it is expected that data loss can be effectively prevented during transmission disruptions, and the stored data can later be used for detailed and precise analysis of radioactive nuclides.

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Keywords: Aerial radiation, Drone, Multi-channel radiation detector, Wireless communication, Radioactive nuclide

Reproducing the Asymmetry Vortex of Historical Typhoons Through Best-Track Analysis

Hyeon-Jeong Kim*, Byung-Il Min, Yoomi Choi, and Ji Yoon Kim

Korea Atomic Energy Research Institute (KAERI)

*liswjd@kaeri.re.kr

Previous studies have reproduced typhoons using the best-track parameters provided by the Joint Typhoon Warning Center (JTWC). Notably, asymmetric vortex has been shown to reproduce storm surges better than symmetric vortex in terms of typhoon reproducibility. Other methods for reproducing typhoons utilize meteorological data from the Regional Data Assimilation and Prediction System, Local Data Assimilation and Prediction System of the Korean Meteorological Agency (KMA), and the meso-scale Model (MSM) of the Japanese Meteorological Agency (JMA). However, these datasets are only available since 2010 for the KMA and since 2002 for the MSM, rendering best-track data essential for reproducing historical typhoons. Nevertheless, the radius by wind speed for each quadrant, which accounts for asymmetric vortex, has only been available since 2001. Hence, a method to estimate the radius of 34-kn wind speeds for data before 2001 is necessary. This study analyzes JTWC best-track data for the period 2002–2015 to estimate the radius of 34 kn wind speeds for typhoons before 2001. Typhoons affecting the Korean Peninsula were selected and categorized into 20–30 °N and 30–40 °N sections to confirm their asymmetry. Typhoon parameters, such as latitude, longitude, maximum sustained wind speed ($V_{\max}$), minimum sea level pressure (P_c), and radius of specified wind intensity for quadrants (RAD), were examined. The correlation between $V_{\max}$ and RAD and the difference between the average RAD and each quadrant were analyzed. Results showed R^2 values of 0.91 (20–30 °N) and 0.80 (30–40 °N). In the 20–30 °N section, the first quadrant was approximately 6% larger than the average RAD, the second and third quadrants were similar to the average, and the fourth quadrant was approximately 7% smaller. In the 30–40 °N group, the first and second quadrants were approximately 4–5% larger than the average, and the third and fourth quadrants were approximately 2–7% smaller. This study applied typhoon asymmetry to historical typhoons, establishing a system to reproduce surges accurately. The findings can be used to calculate hindcast surge heights of specific areas and are expected to contribute to disaster risk assessments, such as hazard mapping and surge frequency analysis.

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Keywords: Best-track, Asymmetry vortex, Quadrant wind speed radius, Maximum sustained wind speed, Storm surge

Numerical Simulation of Coastal Debris Movement With Random Walk Tracking Considering Tidal and Meteorological Effects

Hyeon-Jeong Kim*, Byung-II Min, Yoomi Choi, and Ji Yoon Kim

Korea Atomic Energy Research Institute (KAERI)

*liswjd@kaeri.re.kr

The 2007 Taean oil spill emphasized the need for preemptive disaster prevention measures. Beyond establishing a system for oil spills prediction, it is also vital to establish a system that can predict the movement of floating bodies caused by coastal debris and human casualties. A previous study adopted a two-dimensional finite-element model to reproduce the movement characteristics of floating bodies observed along the Taean coast, producing satisfactory results by incorporating actual tidal and meteorological conditions. However, coastal accidents do not occur only in specific areas, rendering it is necessary to analyze the diffusion tendency according to the tidal and meteorological conditions in the West, South, and East Sea. This study adopted a three-dimensional finite volume model (FVCOM) with water depth integration to represent the complex coastal geometry of the Korean Peninsula for Euler circulation analysis. The Lagrangian passive particle tracking approach was integrated into the model, using a random walk method to simulate movements under coastal hydrodynamic conditions. Tides were modeled as a time series by harmonizing the tidal components of NAO 99b and 99jb at the open boundary. In addition, two meteorological scenarios were considered. The first involved 11 randomly selected days from June 11 to 21, 2024. The second examined the particle behavior over six days, from October 2 to 7, 2016, when Typhoon Chaba affected Korea in 2016, to consider extreme weather conditions. The meso-scale model of the Japanese Meteorological Agency was used for meteorological data. The effects of wind and no wind were analyzed in both scenarios. Particles were released from Incheon and Kunsan on the west coast, Wando and Busan on the south coast, as well as from Pohang and Sokcho on the east coast. Three thousand particles were injected at 100 m intervals in each 3 km section around the port, and 18,000 particles from the six locations were numerically calculated. In the first scenario, the particles moved northward when the northward wind speed reached approximately 13 m/s. In the second scenario, wind directions changed rapidly in each region as the typhoon moved; the particles tended to move in the opposite direction when only tides were considered. The system developed in this study is applicable for real-time forecasting, incorporating both tidal and meteorological factors, and can effectively track coastal debris movement.

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Keywords: Particle tracking, Random-walk, Lagrangian, FVCOM, Extreme weather conditions

Integrating FVCOM With Typhoon Parameter Models for Storm Surge Analysis

Hyeon-Jeong Kim*, Byung-Il Min, Yoomi Choi, and Ji Yoon Kim

Korea Atomic Energy Research Institute (KAERI)

*liswjd@kaeri.re.kr

Previous studies have utilized meteorological data to assess coastal risks due to typhoons. In particular, the meso-scale model (MSM) of the Japan Meteorological Agency (JMA) or the Holland equation, which assumes typhoons as concentric circles, is used to model typhoons impacting the coast of Korea. The Holland equation incorporates the typhoon center location, maximum wind speed, central pressure, and wind reduction factor (WRF). Among these, the WRF, which affects the typhoon's wind speed, has been used to estimate the maximum wind speed (U_{10}) at an altitude of 10 m. According to Powell et al. (2003), U_{10} is approximately 78% (with a standard deviation of 8%) of the mean boundary layer (MBL). Based on this, the WRF is typically applied in the range 0.7–0.9. However, the WRF analyzed in previous studies have been based on hurricanes observed off the coast of the United States. Therefore, it is necessary to perform a sensitivity analysis of the relevant parameters based on typhoons that occur in the Northwest Pacific and affect South Korea. In addition, this study aimed to analyze storm surges using JMA–MSM meteorological data together with the typhoon parameter model. To improve the speed and accuracy of storm surge prediction, this study created an FVCOM grid with a resolution of approximately 300 m, incorporating the finite volume method, to model the Korean coast. The target was typhoons that affected the Korean coast from 2002 to 2021. Equations proposed by previous researchers for calculating the asymmetry and symmetry of the vortex were integrated into the FVCOM model, which utilized typhoon parameters from the best track data of the Joint Typhoon Warning Center. In particular, the typhoon parameter model was subjected to sensitivity analysis using the WRF, and the influence of the WRF on storm surges was analyzed. In addition, the water level data measured at the coastal tide gauge station during the typhoon period were harmonically decomposed using TIRA and MARIE of the TASK-2000 package to calculate the storm surge. The analysis results show that the JMA-MSM meteorological data produced the smallest error in predicting maximum storm surge heights. The WRF sensitivity analysis indicates that changes in wind speed alter the stress on the sea surface, which in turn influences the storm surge height. FVCOM integrating the typhoon parameter model from this study can forecast storm surges.

Acknowledgements

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Keywords: JMA-MSM, Symmetry vertex, Asymmetry vertex, Wind reduction factor, Storm surge

A Methodological Review on the Development of Adsorbents for Extracting Dissolved Resources From Seawater

Ji Eun Park* and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*Jepark12@khnp.co.kr

It is known that terrestrial metal resources are approximately 4.59 million tons, sufficient for about 100 years of use. However, recent surges in metal resource prices and supply instability have highlighted the need for alternative solutions. On the other hand, it has been reported that seawater contains metal resources approximately 1,000 times the amount of terrestrial deposits, which is equivalent to about 60,000 times the global annual demand. However, the average concentration of dissolved metal resources in seawater is about 3.3 ppb, which is extremely low, making it challenging to develop economically viable adsorbents for extraction.

This study explores various methods for developing adsorbents capable of recovering dissolved metal resources from seawater. First, we investigated technologies that activate the adsorbent substrate using radiation or catalysts, followed by the attachment of functional groups that can form coordination bonds with metal resources. Additionally, we examined techniques that maximize adsorption performance using various organic substrates and explored methods of adsorption utilizing light or electricity. Through these approaches, it is expected that the development of economically feasible adsorbents for dissolved metal resources in seawater will be achievable.

Keywords: Metal resources, Adsorbents, Seawater, Radiation, Catalyst

A Case Study on the Radiological Environmental Monitoring Program for Decommissioning of NPP

HyunMin Kim* and Choi Hyu-Chang

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hyunmin.kim@khnp.co.kr

The type of radioactive waste that may occur in the process of nuclear power plant dismantling can be classified into solid, liquid, gas, and mixed waste. The amount of these wastes must be defined in the Final Decommissioning Plan for approval of the licensing. In the case of the Radiological Environmental Monitoring Program (REMP), it is also necessary to identify the nuclides generated in order to obtain approval for licensing and to evaluate the impact on the Environment.

Based on the experience of decommissioning nuclear power plant in United States, we will analyze the contents of the REMP and use it as a reference in establishing a radiation monitoring plan suitable for domestic regulatory conditions. In particular, we would like to 1) compare the differences in nuclides with domestic REMP, and 2) examine whether there are nuclides deleted or added due to reflection of the characteristics of decommissioning.

In this study, we attempted to review radionuclides that must be included in the REMP.

Keywords: Decommissioning, Radiological environmental monitoring, REMP, Environment impact

Real-Time Monitoring and Safety Management System for Worker Protection in Complex Nuclear Decommissioning Environments

MunSu Han*, DongCheol Kim, Minyoung Jung, KeunTeag Park, and Huigyeong Kim

DAON Technology Co., Ltd.

*hanmunsu@naver.com

Decommissioning of nuclear facilities involves the permanent cessation of operations, followed by the dismantling of structures or the removal of radioactive contamination to exclude the facility from regulatory oversight. Successful decommissioning requires minimizing the generation of radioactive waste and carefully managing radiation safety to protect workers involved in the process.

During regular operations, radiation safety measures are controlled by limiting the number of workers and workspaces to reduce exposure. However, during decommissioning, multiple individual work sites may involve cutting and crushing operations, and the loss of power to the fixed HVAC system must be considered. These operations create complex environments where workers' safety can be compromised by the release of radioactive contaminants.

To address these challenges, this study proposes a novel safety management system that integrates real-time radiation monitoring with a mobile air purification system for small-scale work units. This system is expected to improve the efficiency of safety management during nuclear decommissioning.

Acknowledgements

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Keywords: Nuclear decommissioning, Air purification, Real-time monitoring, Worker safety, Radioactive contamination, Modular system

Exploring the Utility of U-234/U-238 Ratio in Groundwater at Gyeongju Disposal Facility for Managing Radiation Exposure to Resident

Seung hyeong Lee*, Sang Hun Lee, Eun-Sang Park, and Min Seong Kim

Korea Radioactive Waste Agency (KORAD)

*lsh9579@korad.or.kr

Currently, a radiological environment impact of Gyeongju disposal facility including the evaluation of radiation exposure due to groundwater intake, is being assessed based on the maximum dose values from nuclide analysis for 8 sites of groundwater testing wells in the disposal facility. However, in reality, unlike other samples, a person who wants to consume groundwater can choose which well to consume.

We studied how U-234/U-238 activity ratio in groundwater could be used as a reference factor to select well to minimize the exposure to surrounding residents to Gyeongju repository.

Many studies have shown that the concentration of U-234 in natural groundwater is higher than that of U-238. The ratio of two uranium isotopes is mainly affected by the age and type of the rock.

In this study, it is assumed that each groundwater collection point has specific geological uniformity if a statistical significance is shown with 5 years of accumulated data of U-234/U-238 activity ratio. In addition, an effective dose evaluation due to H-3, C-14, Sr-90, and uranium radioisotopes was conducted for each groundwater collection point to obtain the correlation between the ratio of U-238/U-234 and the effective dose to determine whether regional restrictions were truly effective.

Result from 1-way ANOVA, of the U-234/U-238 activity ratios for each groundwater well, the p-value was $1.75E-25$. The t-test from any pair of well, EM-1~EM-8, the p-value was below 0.05, except the 2 cases, EM-2, 4 and EM-6, 8 pairs. The correlation coefficient between the effective doses with 5-year average intake for each groundwater well and the U-238/U-234 ratios is 0.70. It can be shown that the lower the U-234/U-238 activity ratio has a trend of the higher the effective dose. If intake from an area with the lowest U-234/U-238 ratio is excluded, the annual effective dose value from groundwater intake decreases by 29%.

In this study, the U-234/U-238 isotope ratios were calculated to give geological uniformity to each groundwater well, and it was found that the effective doses represent a negative correlation with the U-234/U-238 activity ratios. However, the calculated effective dose value from groundwater intake is $8.11E-04$ mSv/yr in 2023 that is only 0.0811% of the public dose limit, 1 mSv/yr. This result shows there should not be required regulatory limit to the intake from groundwater wells. Nonetheless, monitoring whether the U-234/U-238 ratio at each point is within the acceptable statistical range can help the quality control of the site surveillance.

Keywords: Groundwater, Effective dose, Isotope ratio, Uranium

Environmental Survey and Analysis for Hotspot Localization in the Radiation Areas

Giyeon Kim*, Hyun Young Kim, Yewon Kim, Myungsoo Kim, Jaeyeong Jang, and Heejun Chung

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*kimgy@kinac.re.kr

Our research team has developed a gamma probe for rapid and accurate exploration in areas without existing environmental radiation data, aiming to detect leaked nuclear and radioactive materials. As it is distributed ahead of personnel to gather data in these unknown environments, the probe must function independently. Additionally, it must identify hotspots from a distance.

To address these needs, gamma probes were developed using a plastic scintillator and SiPM sensor to make sufficient geometrical detection efficiency and signal. Various analytical methods were applied to the data obtained from the probes. The overall operational procedure of the probe, from measurement to data analysis, is as follows:

- Distribution of multiple probes across a wide area
- Conduct environmental radiation surveys
- Localization of hotspots through statistical analysis of the measurement results.

In this study, the wide area search based on the scenario with leaked nuclear material was primarily conducted to make the overall radioactivity map. The secondary scan was carried out to identify the specific hotspot. Additionally, the second case assuming some probes are malfunctioned was tested and compared with the case of all probes work properly in order to know how many probes actually required for identifying the exact hotspot.

Acknowledgements

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Keywords: Gamma sensor, SiPM, CsI (Tl), Environmental radiation

Study on the Quench Correction Curves for Low Level Tritium Measurement

Seongkyeong Cho^{1,2}, Jong Man Lee^{1,2}, Kyoung Beom Lee², Yeong Jae Park², Dong Hye Heo², Tae Hee Kim²,
and Byoung Chul Kim^{2,*}

¹University of Science and Technology (UST)

²Korea Research Institute of Standards and Science (KRISS)

*bchkim@kriss.re.kr

Tritium (^3H) is primarily produced in nuclear power plants and can combine with oxygen to form tritiated water (HTO). Because of its similarity to regular water, tritium spreads rapidly in the environment and accumulates in living organisms, increasing the risk of radiation exposure. Therefore, accurate and reliable measurement of tritium in such waste is essential for protecting the environment and public health. The recent wastewater discharge from the Fukushima nuclear power plant has further emphasized the importance of detecting and managing tritium.

Liquid scintillation counting (LSC) is an essential tool for measuring low-energy beta emitters like tritium, providing high sensitivity and effective measurement capabilities. However, LSC measurements can be affected by quenching, which reduces the number of detectable photons, thereby weakening the measurement signal. To address this issue, it is necessary to develop and apply quenching correction curves tailored to the specific measurement conditions.

In this study, a quenching standard set composed of nine 100 Bq sources and one background was prepared to generate quenching correction curves. In addition, low-level tritium samples with concentrations of 1,000 Bq/kg, 100 Bq/kg, 50 Bq/kg, 25 Bq/kg, 15 Bq/kg, and 5 Bq/kg were measured to enhance detection accuracy. These concentrations were selected based on the minimum detectable activity (MDA) of 5 Bq/L established by the Nuclear Safety and Security Commission, along with levels commonly observed in practical measurements.

The purpose of this study is to characterize quenching correction curves and use them to accurately measure the minimum detectable concentration of tritium, thereby enhancing measurement reliability. This research is expected to play a critical role in ensuring the safe handling and accurate assessment of tritium in various contexts.

Keywords: Tritium (^3H), Liquid Scintillation Counter (LSC), Quenching correction curve, Minimum Detectable Activity (MDA)

A Study on Self-Absorption and Coincidence Summing Correction Factors for Soil Sample

Ka-Young Park, Jong Man Lee, Kyoungbeom Lee, Tae Hee Kim, Dong-Hye Heo, Yeong Jae Park,
and Byoung-Chul Kim*

Korea Research Institute of Standards and Science (KRISS)

*bchkim@kriss.re.kr

The self-absorption and coincidence summing correction factors are necessary for precisely measuring the activity of radionuclide in environmental material.

In this study, we calculated both the self-absorption and coincidence summing correction factors for soil sample using three Monte Carlo simulation codes and compared the results. We simulated a p-type high purity germanium detector using the Monte Carlo code GEANT, as well as the Monte Carlo based software GESPECOR and LVis. The model was validated by comparing the calculated full-energy peak efficiency with the experimental results, using a standard sample of with a density of 1 g/cm^3 . The correction factors were determined for soil samples in cylindrical bottles with different densities. The elemental composition of the soil sample, determined through the major elements analysis, was used for the simulation.

Keywords: Self-absorption correction factor, Coincidence summing correction factor, High purity germanium detector, Monte carlo simulation, Naturally occurring radioactive material

Gaseous Tritium Light Sources (GTLS) Testing Facility Construction

Sung-Hoon Jung, Seung-chul Shin, and Soon-hwan Son*

AGENCORE CO., LTD.

*shson@agencore.net

The purpose of establishing this testing facility is to verify the performance of tritium self-luminous devices containing tritium by conducting seven types of environmental tests that directly apply stress. The goal is to ensure that the product does not leak radiation. To achieve this, tests are designed based on various conditions, considering the characteristics of the product exposed under different environments. The types and methods of testing are based on standards proposed by the American National Standards Institute (ANSI).

The seven environmental tests are Discoloration test, temperature test, Thermal shock test, Low-pressure test, Drop test, Vibration test, Immersion test. The test levels are determined based on the amount of tritium injected into the product, with conditions ranging from the lowest level 1 to the highest level 4. For performance verification beyond the specified levels, an additional level X test can be performed. Discoloration Tester is set with a temperature range of 40~110°C and a humidity range of 10~100% based on BPT standards. The equipment is designed as a RACK MOUNT type to accommodate all samples and includes adjustable light sources for varying intensities. Temperature Tester is The temperature range is from -65 to 90°C, with a temperature increase and decrease rate set to 3°C per minute to enhance reliability. The equipment is minimized in size considering the sample size. Thermal Shock Tester is Designed with a temperature range of -40 to 150°C, the equipment uses chains and air valves to ensure transfer between high and low-temperature zones within 10 seconds. Low-Pressure Tester is Uses a vacuum pump to create a low-pressure environment inside the testing chamber. The door is equipped with a viewing window for visual inspection and a secondary support to prevent breakage of the window. Drop Tester is Utilizes air valves to automate the dropping and retrieval of samples. A measurement ruler is installed to measure the height of the drop. Vibration Tester is Considering the small size of the samples, a small jig is created to secure the sample. A shaker is used for vibration, and a function generator is employed to adjust frequency output. Immersion Tester is Takes into account the characteristics of moving between low and high temperatures, with separate constant temperature tanks for each. The stainless steel immersion mesh is used to facilitate the testing of small and lightweight samples. Considering the small size and lightweight nature of the products, the equipment has been designed with minimal dimensions. This testing facility will be used to improve the quality of future tritium self-luminous devices and to validate their radiation safety.

Acknowledgements

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Keywords: Tritium, GTLS, Discoloration test, Temperature test, Thermal shock test, Low-pressure test, Drop test, Vibration test, Immersion test

Comparison of Storm Surge Simulations of Identical Typhoons Using Prediction Systems Developed by Different Institutions

Byung-Il Min¹, Hwa-Young Lee², Hyoung-Seong Park³, and Hyeon-Jeong Kim^{1,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Hydrographic and Oceanographic Agency (KHOA)

³National Disaster Management Research Institute (NDMI)

*liswjd@kaeri.re.kr

The Korea Hydrographic and Oceanographic Agency (KHOA) and the National Disaster Management Research Institute (NDMI) have developed systems for predicting storm surges to respond to an average of three typhoons per year that develop in the Northwest Pacific and affect the coast of Korea. KHOA primarily focuses on low-lying coastal areas, including ports, as regions of interest (ROI) to generate inundation maps by setting the typhoon frequency, central pressure, and artificial typhoon tracks. NDMI forecasts the impact of typhoons on the coast using a storm surge prediction system based on typhoon advisories from the Korea Meteorological Administration. The Korea Atomic Energy Research Institute (KAERI) focuses on nuclear power plant (NPP) sites as the ROI to hindcast storm surges and tidal levels caused by typhoon-induced waves and tides and to calculate external forces for detailed evaluations of their impacts on NPPs. The aim of this study was to compare storm surges hindcast using the prediction system of each institution.

KAERI, KHOA, and NDMI implemented ADCIRC, a two-dimensional finite element model with depth integration and an asymmetric vortex equation, to calculate the wind speed radii for each quadrant. Thirty-one typhoons that directly or indirectly affected Korea between 2002 and 2019 were selected, and the best-track data from the Joint Typhoon Warning Center were applied. KAERI applied NAO99j and NAO99jb for open-boundary conditions, whereas KHOA and NDMI used FES2004. The triangular grid resolution was approximately 20 m in NPP regions for KAERI, 300–500 m along the southern coast, approximately 2 km along the western coast, and approximately 5 km along the eastern coast. KHOA's grid resolution was approximately 300–400 m along the southwestern coast and 500 m along the eastern coast, whereas NDMI employed a resolution of approximately 100 m in the eastern, western, and southern coastal areas. The simulation results were compared with the observed peak storm surge heights at 20 tidal observatories, yielding root-mean-square error (RMSE) values of 0.23, 0.20, and 0.18 m for KAERI, KHOA, and NDMI, respectively. Despite the differences in grid resolution and open-boundary conditions, the RMSE differences were relatively small, and all institutions obtained satisfactory results. In the future, cooperation between institutions is expected to continuously compare research results and improve existing models.

Acknowledgements

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Keywords: Storm surge, ADCIRC, Best-track, Hindcasting, Grid resolution

7분과

방사화학 (ORAL)

Opportunities and Challenges in Developing Technology for Uranium Extraction From Seawater

Wook Sohn* and Jieun Park

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*wooksohn@khnp.co.kr

The United Nations predicts that the world's population will grow by about 25% by 2050, and the U.S. Energy Information Administration (EIA) projects that global energy consumption will increase by about 34% as a result. The International Energy Agency (IEA) estimates that electricity will account for 50% of total energy consumption by 2050, up from 20% in 2020. In particular, as computing equipment supporting AI technology—now being widely adopted—acts as an “electricity black hole”, power consumption is expected to rise steeply in the future.

For humanity to thrive, we must minimize the global warming effects of rising energy and electricity demand. A power system focused on renewable energy sources that do not produce CO₂ is essential to achieve this. However, the challenge with renewables is their intermittency, as they are heavily influenced by variable natural conditions like sunlight and wind. This makes nuclear energy crucial. Nuclear power can generate electricity reliably without the issue of intermittency and is the second-largest low-carbon energy source after hydropower.

The main issue with nuclear energy is that uranium, its fuel, is finite, much like fossil fuels. The IAEA estimates there are about 6 million tons of confirmed uranium reserves on land, which would last approximately 100 years. However, if we consider uranium dissolved in seawater, the outlook changes dramatically. The oceans are estimated to hold around 4.5 billion tons of uranium—about 800 times more than land reserves. This vast amount could power 440 reactors for at least 60,000 years. Moreover, since uranium in the ocean is replenished from the Earth's crust as it is extracted, nuclear energy could, in effect, be considered a renewable resource. For this reason, Nature selected seawater uranium extraction technology as one of the seven chemical separation technologies that could change the world.

The challenge lies in economically extracting uranium from seawater, where it exists in very low concentrations as uranyl carbonate complex ions ($[UO_2(CO_3)_3]^{4-}$), at about 3.3 parts per billion (ppb). This concentration is equivalent to the weight of a sugar cube relative to the weight of three fully loaded jumbo jets, making profitable extraction difficult. However, with the price of uranium surpassing \$100 per pound for the first time in nearly 20 years—due in part to Russia's invasion of Ukraine—and expected to continue rising, the economic viability of seawater uranium extraction has gained new momentum.

Given these factors, this paper reviews the current status of R&D in key countries like Japan, the United States, and China, focused on economically extracting uranium from seawater.

Keywords: Uranium, Extraction, Seawater, Adsorbent

Fabrication of Electrospun Nanofibrous Adsorbents for Radioactive Nuclides and Useful Element

YoungWoong Kim¹, Hyunseok Kim¹, Geun Tae Pak¹, Sihyun Kim¹, Tae Hyeon Kim¹, Keun-Woo Lee¹,

Hee-Man Yang², and Taek Seung Lee^{1,*}

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*tslee@cnu.ac.kr

Electrospinning is a method of fabricating nanofiber mats, which are one of the most promising structures as adsorbents with large specific surface areas and high permeability. Electrospinning fibers are commonly referred to as nanofibers but are often sub-micron scale rather than nano scale. This limited the specific surface area of the adsorbent, resulting in a decrease in adsorption capacity. For this reason, an adsorbent with a porous structure was designed.

¹³⁷Cs is a representative radioactive isotope that has a long half-life and emits strong gamma rays. Thus, removal of the radioactive Cs ion has become an important issue, and many efforts are devoted to developing adsorbents for the removal of Cs ions from contaminated aqueous environments. However, radioactive Cs ions are difficult to eliminate because of their low concentrations in nuclear waste, compared with the abundant ions, Na and K ions in seawater. Prussian blue (PB) is one of the well-known adsorbents with high selectivity for Cs ions. To introduce PB on electrospun mesoporous silica nanofibers (MSFs), the MSFs were functionalized with ethylenediamine moiety to bind to Fe (III) ions, which would firmly anchor PB. The maximum Cs ion adsorption capacity using the method of double exposure to Fe (III) ions was determined to be 14.66 mg/g.

⁶⁰Co has a relatively long half-life ($t_{1/2} = 5.27$ years) and emits strong gamma rays (2.5 MeV). Once radioactive ⁶⁰Co is released, it enters the tissues of animals and plants, emitting gamma rays, and causing various radiation-induced diseases to humans. For these reasons, the removal of ⁶⁰Co from radioactive wastewater has been extensively investigated. Most of these adsorbents are fabricated with nano- and microparticles because a large surface area is helpful for efficient adsorption. A Co²⁺ adsorbent was prepared using electrospun porous polyacrylonitrile (PAN) nanofibers, featuring easy recovery for reuse compared with a nanoparticle-based adsorbent. As an efficient ligand for Co²⁺, ethylenediaminetetraacetic acid was introduced on the surface of porous PAN nanofibers with the aid of a branched polyethyleneimine linker to obtain an adsorbent with carboxylic acid groups, making it possible to remove Co²⁺ over a wide pH range. An investigation of the adsorption isotherms and kinetics of the nanofibrous adsorbent indicated that monolayer chemisorption was achieved with a maximum Co²⁺ adsorption capacity of 8.32 mg/g. In addition, radioactive ⁶⁰Co was efficiently removed by the adsorbent with a removal extent of more than 98%.

As the clean energy system becomes more significant, the nuclear energy system is expanding. For this reason, a stable supply of uranium has also become very important. Uranium extraction is demonstrated as an important solution to address this issue using amidoxime known as a particularly promising candidate for uranium extraction due to its unique affinity. The adsorbents were fabricated using spun yarn, nonwoven fabric, and nanofiber web to perform efficient uranium adsorption.

Keywords: Nanofibrous adsorbents, Adsorption, Radioactive nuclide, Uranium

Comparative Effects of Organic and Inorganic Ligands on Uranium Release From Uranophane

Salkenova Zarina and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongun@postech.ac.kr

Uranium (U), vital for global energy production, poses significant environmental challenges due to its potential mobility and bioavailability in ecosystems. While typically found at 1-3 ppm concentrations in the Earth's crust, uranium can also accumulate in organic matter (OM)-rich sediments, reaching levels that may pose environmental risks. Uranium mobility is governed by a complex interplay of biogeochemical factors, including oxidation state, pH, and the presence of organic and/or inorganic ligands.

Uranophane $[\text{Ca}(\text{UO}_2)_2\text{SiO}_3(\text{OH})_2 \cdot 5\text{H}_2\text{O}]$, a key secondary uranium silicate mineral, significantly influences uranium's environmental behavior. It commonly forms in uranium mining deposits and spent nuclear fuel repositories, particularly in groundwater and quartz-rich calcium and silica sediments. Although uranophane is a significant secondary uranium-containing mineral that often forms under oxidizing conditions, the effects of various ligands on its solubility remain insufficiently explored, especially in the presence of both organic and inorganic ligands.

To bridge this knowledge gap, we conducted a comprehensive study on uranophane batch dissolution tests in the presence of selected organic and inorganic ligands. Synthesized uranophane, characterized by XRD analysis, was subjected to dissolution experiments at pH 8 in the presence of organic ligands (humic acid and citric acid at 50, 100, and 150 ppm, respectively) and inorganic ligands (sodium nitrate, chloride, carbonate, and silicate at 10^{-4} - 10^{-6} M). Each experiment utilized 5 mg of uranophane in 45 mL of solution, with uranium release monitored over 60 days using ICP-MS, and pH consistently controlled.

The results highlight the distinct impacts of different ligands on uranophane dissolution. Among the organic ligands, citric acid had the most pronounced effect, increasing uranium concentration to 20.9 ppm at equilibrium, compared to 4.8 ppm with humic acid. This fourfold increase suggests a strong chelation effect due to citric acid's carboxyl and hydroxyl groups, which enhance uranyl ion solubility. For inorganic ligands, increasing carbonate content showed the most pronounced effect on the uranophane dissolution, consistent with their known ability to form stable uranyl-carbonate complexes. The experiment is ongoing, with plans to explore mixed phases of dominant ligands (citric acid + carbonate) to further quantify uranium release, and the results will be included in a final presentation.

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Keywords: Radioactive waste, Uranium, Uranophane, Uranium mobility

Visible-NIR Absorption Spectroscopic Investigation on the Formation of Calcium Neptunyl(VI) Tricarbonate Complex

Hyeongjin Kim¹, Seonggyu Choi², Sangki Cho², Hee-Kyung Kim², Hye-Ryun Cho², and Yongheum Jo^{1,*}

¹Hanyang University

²Korea Atomic Energy Research Institute (KAERI)

*yongheumjo@hanyang.ac.kr

Hexavalent actinides, i.e., actinyl(VI), exhibit a strong affinity for carbonate ions and readily form carbonate complexes. Beyond this carbonate complexation, alkaline earth metals such as calcium and magnesium interact with actinyl tricarbonate species to form alkaline earth metal-actinyl(VI)-carbonate species (i.e., M-An(VI)-CO₃). To date, the formation of M-An(VI)-CO₃ has been proposed only for uranium and plutonium. In particular, calcium uranyl tricarbonate species (i.e., CaUO₂(CO₃)₃²⁻ and Ca₂UO₂(CO₃)₃(aq)) significantly impact the environmental chemistry of uranium in natural waters (pH 7–10, equilibrated with atmospheric CO₂ with environmental level of Ca concentration). Given the chemical similarities within the actinyl (VI) series, it is hypothesized that neptunyl (VI) (NpO₂²⁺) can also form analogous M-Np(VI)-CO₃. However, no experimental evidence for Ca-Np(VI)-CO₃ has been provided yet.

In this context, the present study focuses on investigating the formation of ternary Ca-Np(VI)-CO₃ species using visible–near infrared absorption spectroscopy. The system with NpO₂(CO₃)₂²⁻ and NpO₂(CO₃)₃⁴⁻ in the absence of calcium was first examined by observing the spectral changes of Np(VI) in wavelength range of 550–850 nm with pH varying from 6.5 to 10. From this “pH campaign”, the spectral data of NpO₂(CO₃)₂²⁻ and NpO₂(CO₃)₃⁴⁻ were separately obtained. Subsequently, absorbance measurements under a similar condition to pH campaign, but with varying Ca concentration (up to 1.6 mM) were conducted. The absorption spectra of Np(VI)-CO₃ species exhibited significant changes in the presence of calcium, indicating a chemical shift towards the tricarbonate moiety from NpO₂(CO₃)₂²⁻. This observation aligns with previously reported findings for CaPuO₂(CO₃)₃²⁻. Based on the spectral analysis of the recorded data, the formation of CaNpO₂(CO₃)₃²⁻ was proposed, along with the formation constant of the reaction, $\text{Ca}^{2+} + \text{NpO}_2(\text{CO}_3)_3^{4-} \leftrightarrow \text{CaNpO}_2(\text{CO}_3)_3^{2-}$.

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Keywords: Neptunium(VI), Visible-NIR absorption spectroscopy, Ca-Np(VI)-CO₃ complex

Burnup Determination of HAMP by Chemical Methods

Hyejin Cho*, Namuk Kim, Yang-Soon Park, Minjae Ha, Tae-Hong Park, Hye-Ryun Cho, and Jai Il Park

Korea Atomic Energy Research Institute (KAERI)

*hcho921@kaeri.re.kr

The Ki-Jang Research Reactor (KJRR), currently under construction for the production of medical isotopes and neutron doping, employs plate-type nuclear fuel. The core of these fuel plates consists of low-enriched uranium (LEU) with an enrichment of 19.75wt% U-235, combined with 7wt% molybdenum to form U-7Mo. These U-7Mo fuel particles are dispersed in an aluminum alloy matrix containing 5wt% silicon (Al-5Si), creating a U-7Mo/Al-5Si composite. The cladding of the fuel plates is made from aluminum 6061 alloy (Al-6061). The irradiation testing of the U-7Mo/Al-5Si dispersion fuel, intended for application in the KJRR, was conducted at the High-flux Advanced Neutron Application Reactor (HANARO) at KAERI. This fuel is referred to as HAMP (Hanaro Miniplate fuel).

Burnup is a critical indicator for evaluating fuel performance, efficiency, and reactor safety. Among various measurement methods, the Neodymium-148 (Nd-148) method is widely recognized for its effectiveness. Nd-148 serves as a reliable burnup indicator due to its small cross-section, high fission yield, and comparable fission yields to primary fissile isotopes like ^{235}U , ^{239}Pu , and ^{241}Pu , making it well-suited for accurate burnup assessments.

The Nd-148 method was employed to determine the burnup of HAMP through a precise two-step dissolution process. The dissolution procedure is composed of two-steps for sequential dissolution of cladding material and alloy fuel. The fuel sample can be dissolved completely in 6 M HCl and 14 M HNO_3 - 1 M HF mixture, maintained at 60~90°C for 6 hours under reflux conditions. A two-step separation for both the unspiked and spiked sample solutions enabled us to achieve the pure separation of various fission products, including U, Pu, and Nd. The isotopic compositions of U, Pu, and Nd were determined using a thermal ionization mass spectrometer (TIMS) by the isotope dilution mass spectrometric method (IDMS), which employed triple spikes of ^{233}U , ^{242}Pu , and ^{150}Nd .

By utilizing these validated techniques, the burnup of HAMP is accurately determined. To further validate the burnup values calculated using the Nd-148 method, a comparison was made with burnup values obtained using the Nd (145+146) method.

Keywords: Burnup, Nd monitor method, U-7Mo/Al-5Si dispersion fuel, HAMP, IDMS, TIMS

Study on the Characterization of Colloidal Particles in Groundwater Using LIBD and NTA Techniques

Euo Chang Jung*, Sang-Ho Lee, Wansik Cha, Kyeong Min Lee, Hye-Ryun Cho, and Hee-Kyung Kim

Korea Atomic Energy Research Institute (KAERI)

*ecjung@kaeri.re.kr

Laser-induced breakdown detection (LIBD) and nanoparticle tracking analysis (NTA) were performed to characterize colloidal particles in groundwater. The LIBD method shows excellent detection sensitivity for particles smaller than 50 nm, but it is difficult to distinguish particles by size in polydisperse samples. The NTA method measures the size of each particle individually in polydisperse samples regardless of other particles, but a sufficient number of particles (10^7 – 10^9 particles per mL) must be analyzed for statistically reliable particle size distribution. Thus, these two methods can be used complementarily to obtain information on the size and concentration of trace colloidal particles in groundwater.

Samples were collected from the KAERI Underground Research Tunnel (KURT) site and the tube well (EM-8) installed for environmental radioactivity investigation at the Korea Radioactive Waste Agency (KORAD) in Gyeongju. Using tangential flow ultrafiltration system, 20 L of groundwater was concentrated to 200 mL to facilitate colloidal particle detection. Polystyrene, silica, alumina nanoparticles of known sizes were used as standards for the particle size measurements. The self-built LIBD system adopts the optical detection scheme using a CCD camera to measure the spatial distribution of the breakdown events along the laser beam propagation axis and the breakdown probability as a function of incident laser pulse energy. The second harmonic (wavelength 532 nm, pulse width 6 ns, pulse repetition rate 20 Hz) of the pulsed Nd:YAG laser (Continuum, Surelite II) was used for a laser-induced breakdown. Commercial instrument (Malvern, Nanosight LM10) was used for the NTA measurement.

The colloidal particle sizes measured by the LIBD method were all smaller than 80 nm, and in particular, the particle sizes of the concentrated samples were approximately twice that of the raw groundwater samples. The particle size measured by the NTA method showed a size distribution in the range of 80 to 300 nm, and the number of particles increased significantly in the concentrated samples. The concentrations determined by the number of particles measured were in the order of several hundred ppb for raw groundwater, while concentrations in the concentrated samples increased by 10- to 50-fold, depending on the sample. We discuss the characterization of colloidal particles in groundwater by combining these two analytical methods.

Acknowledgements

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Keywords: Colloidal particles, Groundwater, LIBD, NTA

ANN-Based Pulse Shape Discrimination in KAERI-NDP System

Hwijoon Jeong^{1,2}, Jinhwan Kim², Junhyeok Kim², Byunggun Park², and Kyung Taek Lim^{1,*}

¹Sejong University

²Korea Atomic Energy Research Institute (KAERI)

*kl2548@sejong.ac.kr

All-solid-state lithium-ion batteries are characterized by superior energy density and enhanced safety over conventional liquid-electrolyte-based lithium-ion batteries. The performance and safety of these batteries are critically dependent on the properties of their solid-state electrolytes (SSEs). Inhomogeneities in lithium-ion distribution within SSEs during charge-discharge cycles can induce dendrite formation and material degradation, leading to compromised battery performance, enhanced safety risks, and reduced lifespan. Therefore, real-time analytical methods to monitor lithium-ion behavior and distribution within SSEs are vital for assessing and ensuring safety during battery operation.

Neutron Depth Profiling (NDP) is a non-destructive, real-time method that measures the residual energy of charged particles from neutron capture reactions of light elements (e.g., ⁶Li, ¹⁰B, ¹⁴N). This technique enables depth-based quantitative analysis of lithium-ion distribution within SSEs. Korea Atomic Energy Research Institute (KAERI) established KAERI-NDP system utilizing cold neutrons produced by the HANARO research reactor. Beta and gamma rays emitted from the system's structure lead to background signals, which degrade measurement accuracy as the depth increases. In particular, since background signals cannot be measured and subtracted during real-time analysis, techniques for eliminating background at the signal collection level are essential. The system employs a silicon detector with a sensitive depth of 100 μm . However, this type of detector is limited in reducing background signals through the conventional pulse shape discrimination (PSD) method. To address this limitation, this study proposes a PSD method based on an Artificial Neural Network (ANN) with a combination of a digitizer to acquire pulse signals from the preamplifier.

A binary classification model was developed using a multilayer perceptron architecture to classify signals based on input pulse signals. The training dataset was created by distinguishing between charged particles and background signals, incorporating a noise addition technique based on random numbers to enhance the robustness of the dataset. The ANN model demonstrated strong agreement between the predicted and actual background spectra in the energy region above 500 keV and accurately predicted the net count spectrum of charged particles. Furthermore, the model removed background signals, enabling depth analysis of LLZTO up to 23 μm . Therefore, the proposed ANN-based discrimination method has proven effective in eliminating background signals in the real-time operation of the KAERI-NDP system, thereby enhancing the accuracy and efficiency of SSE analysis.

Acknowledgements

This work was supported by the Korean government (MSIT) (1711173768).

Keywords: Solid State Electrolyte (SSE), Neutron Depth Profiling (NDP), Pulse Shape Discrimination (PSD), Artificial Neural Network (ANN)

7분과

방사화학 (POSTER)

Study of Radiochemical Separation of Pu and Am Isotopes in Metal Samples With Anion Exchange Resin and TRU Resin

Myung Ho Lee^{1,*}, Hee Seung Lim¹, Young Jae Maeng¹, Chang Hoon Lee¹, and Myeong Pil Kim²

¹Korea Reactor Integrity Surveillance Technology (KRIST)

²Korea Standards Analysis (KOSA)

*myholee@krist.co.kr

This study presents a rapid radiochemical separation method for Pu and Am isotopes in metal samples with anion exchange resin and TRU resin. After radionuclides were leached from the metal samples with concentrated HCl and HNO₃, the sample was allowed to evaporate to dryness after filtering the leaching solution with 0.45 micron filter. Before Pu analysis, Pu oxidation states were adjusted with NaNO₂ and H₂O₂ reagents in HNO₃ and HCl medium, respectively. The Pu isotopes were adsorbed on anion exchange column in HNO₃ or HCl medium. The Pu isotopes adsorbed on anion exchange column were eluted with several strip reagents. After the passing solutions from anion exchange column were dryness, the Am isotopes were purified with TRU resin. The purified Pu and Am isotopes were measured by alpha spectrometer, respectively, after micro-precipitation of neodymium. The optimized radiochemical separation method for Pu and Am isotopes were applied to high radioactive metal materials samples such as surveillance and baffle-former-bolt samples and determined activity concentrations and activity ratios of Pu and Am isotopes with alpha spectrometry and ICP-MS system.

Acknowledgements

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Keywords: Radiochemical separation method, Pu/Am isotopes, Surveillance and baffle-former-bolt samples, Alpha spectrometer, ICP-MS system

Electrochemical Approaches to the Study of Metal Redox Reactions in Molten Hydroxide Mixtures

Suhyeon Lee, Hyunjin Boo, Suhyun Kim, Sunkyung Park, Tae Hee Kim, and Byung Gi Park*

Soonchunhyang University

*byunggi@sch.ac.kr

Metal contamination exists in the form of an oxide film on the surface. This fixed contamination can be decontaminated by chemical dissolution of the oxide film or by surface scrubbing. In many cases, the decontamination process is performed by an acid-based solution. However, the problem is liquid waste is generated. Electrochemical decontamination processes using hydroxide-based salts are expected to be able to continue to be used without generating liquid waste, although an investigation of electrochemical behavior is necessary to determine their suitability. A mixed salt of sodium hydroxide and potassium hydroxide was used in the experiment because it has a low melting point of 443 K when mixed at 51:49 mole percent. The electrochemical cell was an alumina crucible with three electrodes. Platinum wires were used as working and quasi-reference electrodes, and glassy carbon rod was used as a counter electrode. Cyclic voltammetry (CV) and square wave voltammetry (SWV) were used as electrochemical measurement methods. All experiments were performed at 473 K. CV was measured in 6 steps from 100 mV to 1,000 mV and SWV was measured in 5 steps from 10 Hz to 100 Hz. Electrochemical measurements of NaOH-KOH (51:49 mol %) were performed to confirm the basic reaction against blank salt. Then, Co(OH)_2 was added in steps of 0.3, 0.6, and 0.9wt% and measured CV and SWV, separately. A reduction limit due to reduction to H_2O was observed around -0.9 V (reaction: $2\text{H}_2\text{O} + 2\text{e}^- = \text{H}_2(\text{g}) + 2\text{OH}^-$). An anodic limit was also observed around 0.5 V (reaction: $2\text{OH}^- = 0.5\text{O}_2(\text{g}) + \text{H}_2\text{O} + 2\text{e}^-$). In the electrochemical data measured after the cobalt hydroxide was added to the base salt, a reduction peak around -0.7 V was identified in addition to the previously identified peak. The SWV results confirm that two electrons have been transferred. To confirm that this peak was cobalt hydroxide, electrochemical experiments were performed with the same electrode area by adding cobalt hydroxide to the same solution. As the concentration increased, the value of the current peak increased consistently.

Acknowledgements

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Keywords: Electrochemical decontamination, Cyclic voltammetry, Square wave voltammetry, NaOH-KOH

Chemical and Electrochemical Pretreatment of Radioactive Elements for Analysis of Radioactive Waste

Sang-Eun Bae*, Chan-Yong Jung, Jeong Eun Kim, Hwakyung Jeong, and Junhyuck Kim

Korea Atomic Energy Research Institute (KAERI)

*sebae@kaeri.re.kr

Growing concerns related to radionuclides derived from nuclear activities have spurred the development of efficient separation and disposal technologies for the radionuclides.

To properly dispose of radioactive waste, it is essential to assess its radioactivity. The concentration of radionuclides that constitute at least 95% of the total radioactivity in the waste must be determined by chemical or radioactivity measurement methods before the waste will be sent to a disposal facility. Notably, 14 radionuclides (H-3, C-14, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Tc-99, I-129, Cs-137, Ce-144, and pre-alpha) are of particular interest due to legal disposal requirements.

In this work, we have developed a technology for pretreating a radionuclide such as iodine-129. Traditionally, in order to analyze iodine radionuclide compound, several pretreatment steps such as ion-exchange resin treatment, stirring, solvent extraction, etc. have been required. However, these methods have drawbacks including being time-consuming, costly, generating large amounts of secondary waste, and having poor removal efficiency. Furthermore, for beta radioactivity measurement, tritium needs to be separated from the iodine sample due to their similar beta energies.

Here, we employed selective adsorbents for pretreatment of iodine wastes. The iodine compound could be concentrated by a factor of 100 and then analyzed using UV-Vis spectroscopy, ICP-mass spectroscopy, and liquid scintillation counting. The results showed that the adsorbents can separate iodine species such as iodine, iodide, and methyl iodide from water. This results indicate that the Minimum Detectable Activity (MDA) for the iodine species can be reduced 100 times.

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Keywords: Iodine-129, Radioactivity, Radioactive wastes, Liquid scintillation counter, Separation, Radionuclide analysis

Development of a Separation Method for Metallic Radionuclides in Low and Intermediate Level Radioactive Waste

Jieun Choi*, Soo Bin Park, and Seungil Park

Orbitech Co., Ltd.

*jechoi@orbitech.co.kr

In the low- and intermediate-level radioactive waste generated during the operation of nuclear power plants, various metallic radionuclides are present. Many of these are low-energy gamma and beta emitters, which are difficult to detect. Identifying these hard-to-analyze radionuclides is crucial in the disposal process of radioactive waste, and chemical pretreatment is essential for their analysis.

In this study, we successfully developed a separation method for analyzing metallic beta-emitting radionuclides in radioactive waste based on ion exchange resin techniques. Radioactive Fe, Ni, Sr, and Tc were sequentially separated from wood and polymer samples using coprecipitation and ion exchange resins. The radioactivity of each nuclide was measured using a liquid scintillation counter (LSC). Additionally, to correct for losses during the chemical separation process, the recovery rates were calculated by measuring changes in stable isotope concentrations using ICP-MS.

This separation method may allow for more accurate radioactivity measurements in radioactive waste and provide shorter analysis times compared to conventional methods.

Acknowledgements

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Keywords: Radioactive waste, Radionuclides, Chemical separation, Radioactivity measurements

Thermal Oxidation of Non-Stoichiometric Uranium Oxides and Decomposition of Uranyl Nitrate Hexahydrate at High Temperatures

Hyeonjin Eun¹, Hyelee Kim², Kyeongtae Park³, Hyeongjin Kim⁴, Yongheum Jo⁴, Kwangheon Park⁵,

Won Pyo Jeong⁶, Seungyeon Choi⁶, and Jaeyeong Park^{3,*}

¹Korea Advanced Institute of Science and Technology (KAIST)

²Jeju National University (JNU)

³Ulsan National Institute of Science and Technology (UNIST)

⁴Hanyang University

⁵Kyung Hee University

⁶Korea Nuclear International Cooperation Foundation (KONICOF)

*jypark@unist.ac.kr

Understanding the physical and chemical properties of solid-state uranium is crucial in the nuclear fuel cycle. Uranium oxides are non-stoichiometric solids and exist in various oxidation states. Uranium oxides with different O/U ratios occur between tetravalent UO_2 and hexavalent UO_3 , such as U_4O_9 ($= \text{UO}_{2.25}$), U_2O_5 ($= \text{UO}_{2.50}$), U_3O_7 ($= \text{UO}_{2.33}$), and U_3O_8 ($= \text{UO}_{2.67}$). Among these, hyperstoichiometric uranium dioxide, which maintains a cubic structure, is referred to as UO_{2+x} ($0 \leq x \leq 0.25$). In this study, the value of x of an unknown UO_{2+x} sample was determined by using the lattice parameter and a weight change during thermal oxidation to U_3O_8 .

Uranyl nitrate hexahydrate ($\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$) and uranium dioxide (UO_2) were heated at 400°C, 450°C, 500°C, and 550°C, each for 30 minutes. The reactions were analyzed by comparing weight differences, color changes, and X-ray diffraction (XRD) analysis. Uranyl nitrate hexahydrate, a hexavalent uranium salt, converts to UO_3 as water and nitrate are removed upon heating. Further heat treatment results in thermal decomposition, forming U_3O_8 . On the other hand, UO_2 showed its direct oxidation into U_3O_8 during heat treatment. Thermogravimetry-differential thermal analysis (TG-DTA) was performed on uranyl nitrate hexahydrate and the unknown sample UO_{2+x} . Uranyl nitrate hexahydrate exhibited four water peaks and two nitrate peaks, while UO_{2+x} did not exhibit any clear peaks in the DTA curve. A 2.55wt% weight increase in UO_{2+x} , leading to the formation of U_3O_8 , indicates that $x = 0.23$. This value is identical to the x value of 0.23 ($a = 5.443 \text{ \AA}$) obtained from the linear interpolation of the lattice parameters between UO_2 ($a = 5.470 \text{ \AA}$) and U_4O_9 ($a = 5.441 \text{ \AA}$) through XRD analysis.

Even when UO_2 and UO_{2+x} are heated under an atmosphere with 21% oxygen, it is not easy to oxidize them to UO_3 . This is because the activation energy required to alter the stable fluorite structure is high, resulting in prolonged reaction kinetics. Even if oxidation occurs, stable U_3O_8 is formed, preventing further oxidation. However, since the nitrate form is already hexavalent, UO_3 can be formed without an oxidation reaction, and the nitrate form itself is highly reactive with oxygen and less stable, allowing the formation of UO_3 .

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Keywords: Thermal oxidation, Thermal decomposition, Non-stoichiometric uranium oxides, TG-DTA, XRD

Validation of Sr-90 Separation and Quantification by Carbonate Precipitation in Radioactive Waste

YunHan Lee*, DongGyu Lee, JunHwi Bang, Chang Ryong Lee, Heung Nae Lee, and JeongSoo Kim

Korea Standards Analysis (KOSA)

*yunhan.lee@ko-sa.co.kr

In order to validate the method for separating and recovering Sr-90 nuclide from radioactive waste, it is necessary to establish a validation method and establish evaluation criteria for six validation items to ensure the reliability of the Sr-90 separation and quantification method. The six validation items are selectivity, linearity, accuracy, precision, MDA calculation, and uncertainty. The experiment was performed by adding Sr-90 RM and interfering nuclides (such as Fe-55, Ni-63, Co-6, etc.) to simulated samples. The selectivity of the six validation items was verified by measuring before and after the interfering nuclide separation, the linearity was verified by measuring the diluted sample by LSC by concentration, the accuracy was verified based on the average value of the measured radioactivity value compared to the radioactivity value of the added standard, the precision was verified for repeatability and reproducibility, the MDA was calculated using an equation, and the measurement uncertainty was evaluated using uncertainty guidelines. The results of the above six validation items all satisfied.

Keywords: Separation, Validation, Sr-90

Facile Hydrothermal Synthesis of Magnetic Prussian Blue/Maghemite Composite for Selective Cesium Ion Removal

Hye Ran Noh^{1,2}, Youngjin Cho^{1,3,*}, Dong-Woo Lee^{1,2}, Kyung-Tae Ko⁴, Tae-Hyeong Kim¹, Jeongmook Lee^{1,2}, Jong-Yun Kim^{1,2}, Tae-Hong Park^{1,2,*}, and Sang Ho Lim^{1,2,*}

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

³Korea Food Research Institute (KFRI)

⁴Korea Basic Science Institute (KBSI)

*slim@kaeri.re.kr

The presence of radioactive cesium species (e.g., ^{137}Cs) in radioactive waste have necessitated the development of effective strategies for their selective removal from contaminated water. In this study, we synthesized nickel-modified Prussian blue/maghemite (NiPB/ $\gamma\text{-Fe}_2\text{O}_3$) composite material, which exhibit high selectivity and efficiency for cesium ion removal. The composite was synthesized via a straightforward hydrothermal method, where NiPB was dispersed in deionized water and subjected to hydrothermal treatment at 140°C for 4 hours. The resulting composite demonstrated magnetic properties that facilitate its easy separation from cesium-contaminated solutions. The magnetic origin of the composite was investigated through X-ray absorption spectroscopy (XAS) and X-ray magnetic circular dichroism (XMCD) measurements. XAS analysis revealed significant changes in the Fe L-edge spectra after hydrothermal treatment, indicating partial oxidation of Fe^{2+} to Fe^{3+} and the formation of $\gamma\text{-Fe}_2\text{O}_3$. Despite the lower oxide content relative to NiPB, the XMCD measurements clearly showed that the magnetism in the composite originated from $\gamma\text{-Fe}_2\text{O}_3$. The NiPB/ $\gamma\text{-Fe}_2\text{O}_3$ composite exhibited a cesium ion removal efficiency of $\geq 99.7\%$ in solutions containing various alkali and alkaline earth metal cations. The maximum adsorption capacity was determined to be 102 mg g^{-1} using the Langmuir adsorption model. These findings underscore the potential of NiPB/ $\gamma\text{-Fe}_2\text{O}_3$ composites as a promising material for selective cesium ion removal, offering a viable solution for environmental remediation and radioactive waste management.

Keywords: Prussian blue analogue/ $\gamma\text{-Fe}_2\text{O}_3$ composite, Magnetism, Cesium removal

Raman and FTIR Spectroscopic Studies of Uranophane and Sklodowskite as Solubility-limiting Solid Phases in Oxidic Groundwater

Wansik Cha*, Hye-Ryun Cho, Hee-Kyung Kim, and Eui Chang Jung

Korea Atomic Energy Research Institute (KAERI)

*wscha@kaeri.re.kr

Understanding the dissolution behavior of stable mineral phases containing radionuclides (RNs) is crucial for accurately assessing the long-term safety of deep geological repository systems. The initial step in elucidating the migration behaviors of RNs is to identify the relevant solubility-limiting solid phases (SLSPs) among the various naturally occurring mineral forms. In previous studies, we successfully synthesized two naturally occurring Ca- and Mg-U(VI)-silicates, uranophane and sklodowskite, in the laboratory. Their structural identities were confirmed through x-ray diffraction patterns, scanning electron microscopy, and electron dispersive spectroscopy. Geochemical modeling indicated that uranophane is the SLSP in the granitic groundwater from the KAERI Underground Research Tunnel.

In this study, we further characterized these synthetic minerals using Raman and FTIR spectroscopy to examine their structural differences. Both minerals share a U:Si ratio of 1:1 in their chemical compositions, with chemical formula of $\text{Ca}(\text{UO}_2\text{SiO}_3\text{OH})_2 \cdot 5\text{H}_2\text{O}$ (uranophane) and $\text{Mg}(\text{UO}_2\text{SiO}_3\text{OH})_2 \cdot 6\text{H}_2\text{O}$ (sklodowskite). The UO_2^{2+} and SiO_4^{4-} ions, along with structural water molecules surrounding Ca^{2+} and Mg^{2+} ions, are detectable molecular units in vibrational spectroscopy. All solid phases were freeze-dried overnight after being collected from the aqueous phase, both before and after long-term (> 3 months) solubility tests in simulated groundwater solutions and 0.1 M NaClO_4 . Each solid sample powder was directly positioned on the surface of a diamond crystal of an attenuated-total-reflection (ATR) module for FTIR spectroscopy and on a Si wafer for Raman spectroscopy using a 633-nm excitation laser.

The measured spectra exhibited characteristic vibrational bands of uranophane and sklodowskite, which were distinct in both Raman and FTIR spectra. The ν_1 and ν_3 vibrational modes of UO_2^{2+} and SiO_4^{4-} ions appeared within the 700–1,100 cm^{-1} region, while the vibrational bands for water molecules were observed in the 1,400–1,800 cm^{-1} and 2,600–3,800 cm^{-1} regions. This spectroscopic investigation was instrumental in detecting any solid-phase changes during the dissolution process. In simulated groundwater, both mineral phases remained stable, whereas in 0.1 M NaClO_4 solutions, their alteration into other solid phases was observed. Additionally, spectral deconvolution analysis of the standard Raman and FTIR spectra of both minerals enabled the isolation of peak properties of individual vibrational bands, which are summarized in a table.

Acknowledgements

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Keywords: Calcium/magnesium uranyl silicate, Uranophane, Sklodowskite, Solubility, FTIR/Raman spectroscopy, Spectral deconvolution

~~Validity Verification for ¹²⁹I Separation and Quantitative Methods~~

~~DongGyu Lee*, HyoWon Kim, KwyHwan Cho, KyoungHo Han, JeongSoo Kim, and KeChon Choi~~

~~Korea Standards Analysis (KOSA)~~

~~*donggyu.lee@ko-sa.co.kr~~

Withdrawal

Shielded Window Maintenance of KAERI Chemical Hot Cell

Min-jae Ha*, Hyung Kwon Lee, Hye-Ryun Cho, and Jai Il Park

Korea Atomic Energy Research Institute (KAERI)

*hmj@kaeri.re.kr

KAERI operates the only chemical hot cell in Korea for nuclear fuel burn-up analysis. We perform regular inspection and maintenance of the chemical hot cell facility to ensure stable experiments. In addition to high radioactivity various types of acids are handled inside the cell for sample dissolution. This can reduce the visibility through the cell's shielded window, affecting the safe sample handling.

In this report, we describe improvement of the shielded window visibility through two in-house maintenance methods of the chemical hot cell.

Keywords: Chemical hot cell, Shielded window, Maintenance

Separation of Niobium in Radioactive Waste Samples by Precipitation of Iron(III) Hydroxide

Kwang Soon Choi^{1,*}, Kyunghun Jung², Jungbo Yoo², Seung-chul Park¹, Heejin Ahn¹, and Mee Suk Song¹

¹Green Radiation Co., Ltd.

²Korea Atomic Energy Research Institute (KAERI)

*kschoi@greenrad.kr

To permanently dispose of radioactive waste, the concentration of at least 25 nuclides, including Nb-94, must be measured. Nb-94 is a gamma-ray-emitting nuclide, so it can be directly measured by gamma spectrometry after dissolution. If Co-60 coexists in the sample solution, the MDA of Nb-94 due to Compton scattering is increased. Therefore, if the MDA of Nb-94 is higher than 1.11 Bq/g (1% of the low-level waste disposal concentration limit), it should be separated before measurement. A sequential separation method for non-volatile beta-ray-emitting nuclides such as Fe-55, Ni-59, Ni-63, Sr-90, Tc-99, including Nb-94, has been established for assessing nuclide inventories. However, when reanalyzing Nb-94, the sequential separation method currently in use is inefficient, so a method for separating only Nb-94 was developed. By adjusting the pH of the solution to 4.5-5 using ammonia, iron precipitates including niobium to form $[\text{Fe}(\text{OH})_3 + \text{Nb}]$, while cobalt remains in the supernatant, allowing for the separation of niobium (Nb) from cobalt. After centrifuging the iron precipitate and supernatant, the precipitate is washed with a pH 4.5-5 solution, increasing the amount of cobalt in the solution and confirming that more than 99.5% of niobium remains in the precipitate. Therefore, the precipitate $[\text{Fe}(\text{OH})_3 + \text{Nb}]$ is dissolved in 3-4 M HCl and then measured by gamma spectrometry to confirm that the MDA of Nb-94 is less than 1.11 Bq/g. If the MDA of Nb-94 is higher than 1.11 Bq/g, it is evaporated and dissolved in 0.1 M HCl/1 M HF and used as the loading solution. Niobium can be separated by combining with the resin in the separation column in a medium of 0.1 M HCl/1 M HF, remaining in the separation column, while cobalt passes through the separation column. The niobium left in the separation column, combined with the resin, is then eluted with 5 M HNO_3 /0.2 M HF, separated, and then measured by gamma spectrometry.

Keywords: Niobium, Iron(III) hydroxide, Precipitation, Anion exchange chromatography, Radioactive waste

Disadvantages of the Sequential Separation for Nonvolatile DTM Nuclides

Kwang Soon Choi*, Seung-chul Park, Heejin Ahn, and Mee Suk Song

Green Radiation Co., Ltd.

*kschoi@greenrad.kr

To permanently dispose of radioactive waste, a minimum of 25 nuclide concentrations must be measured. The Fe-55, Ni-59, Ni-63, Sr-90, and Tc-99 nuclides specified in the Nuclear Safety and Security Commission Ordinance are beta-emitting nuclides that must be separated before measurement. According to the literature, a sequential separation method for six nuclides has been established for inventory assessment. The five elements are sequentially separated in the order of Tc-Sr-Ni-Fe-Nb. Iron, which remains in the separation column combined with the anion exchange resin in the NH_4^+ -oxalate medium, is eluted with 4 M HF for separation. Cobalt is eluted with iron, so the precipitation method and anion exchange chromatography are applied to separate iron and cobalt. This method does not change the solution medium to sequentially separate six nuclides, so it has the advantage of quickly separating six nuclides. However, since iron binds with the anion exchange resin, this method cannot be applied if the amount of iron exceeds a certain amount. In addition, since cobalt is also eluted with iron, the separation process of cobalt and iron has a complicated disadvantage because Co-60 coexists when Fe-55 is measured by LSC. Therefore, it is necessary to develop a sequential separation method to measure the radioactivity concentration of six nuclides in samples with an amount of iron exceeding 50 mg.

Keywords: Sequential separation, Anion exchange chromatography, Radioactive waste

Comparative Analysis of Treatment Efficiency Based on Conditions of Radioactive Liquid Waste High-Speed Treatment System Using Nano-Catalyst

So-Yun Jeong¹, Hyun-Young Shin¹, Seung-Joo Lim², Wook-Jae Yoo¹, Chan-Hee Park¹, Bo-Bae Cho¹,
Sung-Ryul Kim¹, Jung-Min Oh¹, Ba-Ro Lee¹, Ji-Ung Kim¹, Hye-Jin Kim¹, and Kyu-Tae Park^{1,*}

¹Orbitech Co., Ltd.

²Korea Atomic Energy Research Institute (KAERI)

*paradoxno1@orbitech.co.kr

In this study, we developed equipment capable of rapidly processing non-degradable radioactive liquid waste containing an organic complexing agent generated during the overhaul (O/H) of nuclear power plants, involving the replacement of aged equipment and performance enhancement tasks. To evaluate the equipment's performance and derive optimal processing conditions, a simulated waste solution consisting of oxalic acid and transition metal oxides of copper (Cu) and nickel (Ni) was utilized. The removal efficiency of oxalic acid and transition metal oxide in the simulated waste solution for performance verification was derived by performing total organic carbon (TOC) analysis and inductively coupled plasma spectroscopy (ICP-OES).

The concentration of oxalic acid was fixed at 20 mM, while the concentrations of transition metal oxides, sodium persulfate (PDS), and barium hydroxide were varied. The experimental variables for transition metal oxide concentrations were set at 2 mM and 20 mM; sodium persulfate concentrations were 2 mM, 10 mM, 20 mM, 30 mM, and 60 mM; and barium hydroxide concentrations were adjusted to 0.25x, 0.4x, 0.5x, 0.6x, and 0.7x of the simulated waste solution volume. ICP-OES analysis results confirmed that more than 99% of radioactive nuclides were removed under all conditions, and TOC analysis results confirmed that approximately 94% of organic matter was removed under most conditions.

Additional experiments are being conducted by further subdividing the condition variables to maintain the processing efficiency at least 98%. In addition, a commercial design plan is being developed based on the problems and supplementary items of the pilot facility.

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Keywords: Radioactive liquid waste, Transition metal oxide, Nanocatalyst, Oxalic acid

Method Validation of Iodine-129 Radiochemical Analysis in Metal Samples

Kwangeun Lee, Sora Choi, Jung Bo Yoo, and Mun Ja Kang*

Korea Atomic Energy Research Institute (KAERI)

*munkang@kaeri.re.kr

Iodine-129 is beta emitting radionuclide with a maximum energy of 154 keV. It also emits gamma ray at 39.6 keV and X-rays at 29.8 and 33.6 keV. These radiological properties allow for the measurement of radioactivity of iodine-129 using liquid scintillation counting or low energy gamma spectroscopy. Because iodine-129 has a very long half-life of 1.57×10^7 years and high mass equivalent to specific activity, it can be measured by ICP-MS or AMS additionally. In our study, we have determined the radioactivity of I-129 in contaminated solid waste or soil. This method involves leaching of solid sample in oxidant solution, extraction with CHCl_3 , back-extraction with water, precipitation of AgI and X-rays counting with LEGe. We tested and evaluated the metal samples such as SUS 304 and Inconel 600 by this method.

For the method validation, the metal samples were treated with leaching agents of 1.3~1.6% NaClO solution. The dissolved elements from metals were determined using ICP-OES. The 0.33 mg/L of Fe, 0.038 mg/L of Ni, 0.202 mg/L of Cr, 0.102 mg/L of Cu are dissolved from 5 g of SUS 304 in the leaching agents. 0.034 mg/L of Ni, 0.022 mg/L of Cr, 0.014 mg/L of Cu are dissolved from 10 g of Inconel 600. This indicates that the dissolved metal elements in the oxidant leaching agents is negligible.

In this study, SUS 304 and Inconel 600 with different weights were tested by adding I-129 tracer of 46.1 Bq and KI carrier of 107 mg. In the case of SUS 304, the chemical recovery of KI carrier ranges from 0.84 to 0.92. The average and standard deviation are 0.87 and 0.030, respectively. The radioactivities of I-129 tracer measured after leaching in oxidant, extraction and back-extraction, and precipitation, were 45.8, 43.5, and 46.4 Bq for the three samples. Comparing the initial radioactivity of I-129, the ratio of final radioactivity to initial one is evaluated to be between 0.94 to 1.01. In the case of Inconel 600 samples, The range of chemical recovery of KI carrier is 0.87~0.93. the radioactivity ratio is evaluated to be 0.93~1.00.

In conclusion, the radiochemical separation and measurement procedure of I-129 in contaminated solid waste can be adaptable for metal samples. The chemical recovery is over 84% and the ratio of final radioactivity to initial one is evaluated to be between 0.93 and 1.01.

Acknowledgements

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Keywords: Iodine-129, Metal, LEGe, Method validation

Evaluation of Pre-Treatment Process for the Determination of Radionuclides in Soil Waste

Jung-Weon Choi and Jung Bo Yoo*

Korea Atomic Energy Research Institute (KAERI)

*jwchoi92@kaeri.re.kr

A large amount of soil waste is generated in nuclear facilities and is classified as radioactive waste. In order to dispose of the waste, it is required the identification of radionuclides, which constitute 95% of the total radioactivity in the soil waste package. Among them, alpha and beta nuclides require pre-treatment solubilization and chemical separation for each nuclide analysis. In the case of gamma nuclides, the soil itself can be analyzed or the solubilized soil can be used for analysis. Therefore, in order to validate the pre-treatment of soil waste, it is necessary to prepare simulated test soils using certified standard reference soil, spiking them with the elements to be tested in some cases, and solubilizing them with various acids.

In this study, 1 g of the certified Standard Reference Material® 2711a sold by NIST was used to validate the pre-treatment solubilization. The pre-treatment solubilization was carried out by leaching method using aqua regia and hydrofluoric acid at 200°C for 1 hour. The recovery chemical yield of 9 elements (Co, Ce, Cs, Nb, Re, Sr, U, Fe, Ni) was determined using ICP-OES and ICP-MS, proving that soil pre-treatment solubilization using the leaching method is feasible.

The application of developed leaching test conditions to soil waste analysis has the advantage of fast and simple sample preparation. However, because this method uses a small amount of sample, it may be difficult to meet the conditions for acceptance of a domestic repository for some nuclides with low disposal concentration limits. Therefore, we plan to optimise the sample preparation experimental conditions for large sample volumes in the future.

Acknowledgements

This work was supported by the KAERI Institutional Program (Project No. 522320-24).

Keywords: Soil, Validation, Pre-treatment, Radionuclides, Radiochemistry, Acid leaching

Sequential Separation Method for H-3 and C-14 Isotopes Using a Distillation Apparatus

Hee Seung Lim*, Myung Ho Lee, Young Jae Maeng, and Chang Hoon Lee

Korea Reactor Integrity Surveillance Technology (KRIST)

*hslim@krist.co.kr

In nuclear power plants, the accurate measurement of low-level radioactive isotopes, such as H-3 and C-14, is essential for environmental monitoring and safety assessments. This study describes a sequential separation method for H-3 and C-14 isotopes using a distillation apparatus. In a 200 mL sample solution, 200 mg of potassium permanganate and 500 mg of sodium hydroxide were added. The solution was heated, and the first 10 mL of distillate was discarded. The next 10 mL of distillate was collected and transferred to a vial, followed by the addition of 10 mL Ultima Gold LLT to prepare the H-3 measurement sample. The remaining sample was treated with 5 g of potassium persulfate, 2 mL of 1M silver nitrate, and 25 mL of concentrated nitric acid, and then heated for 90 minutes for collecting C-14. To increase the collection efficiency, two vials containing 5 mL of Carbo-Sorb were connected in series to the distillation apparatus to adsorb C-14 isotopes. The Carbo-Sorb solution adsorbed with C-14 was transferred to a vial, and 10 mL of Permafluor was added to prepare the C-14 measurement sample. With Single DPM mode, the prepared H-3 and C-14 measurement samples were measured by the liquid scintillation counter after quenching correction curves. The average recoveries of H-3 and C-14 with CRM were measured to be 99% and 96%, respectively. The sequential separation method for H-3 and C-14 investigated in this study was applied to wasted activated charcoal filter produced from nuclear power plants after validating the reliability by result of proficiency test organized by KRISS.

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Keywords: Sequential separation, H-3, C-14, Distillation method, Wasted activated charcoal filter, Proficiency test

DFT-Enhanced Deep Learning Model for Predicting Ion Interaction Coefficients of Specific Ion Interaction Theory

Sangki Cho* and Hye-Ryun Cho

Korea Atomic Energy Research Institute (KAERI)

*skcho@kaeri.re.kr

The Specific Ion Interaction Theory (SIT) is widely used to calculate activity coefficients. The interaction coefficient (ϵ) between ionic species, a key parameter in the SIT model, is primarily derived from experimental data. However, the current limited database hinders the accurate corrections of all species participating in reactions at high ionic strength. This study presents a novel approach to estimating ϵ values in SIT using artificial neural network (ANN) techniques combined with physical features derived from Density Functional Theory (DFT). DFT is applied to generate relevant atomic and molecular features for individual ions and ion pairs, which are then fed into ANN architecture to predict ϵ values. The model was trained on a comprehensive dataset of experimentally determined ϵ values and corresponding DFT-calculated features, showing significant improvements in prediction accuracy compared to methods that rely solely on ion charge information.

The present DFT-ANN hybrid model demonstrates improved performance in capturing the complex interactions between ions, providing accurate ϵ values across a wide range of ionic species. By utilizing DFT-derived features, this model captures subtle electronic and structural properties of ions that influence their interactions, going beyond the limitations of charge-based predictions. This approach not only enhances the precision of estimated ϵ but also provides a framework for integrating quantum mechanical insights with data-driven techniques in physical chemistry. The results demonstrate the potential of this hybrid method for advancing the understanding and application of ion-specific effects in complex chemical systems, particularly where experimental data is limited.

Keywords: Specific Ion Interaction Theory (SIT), Density Functional Theory (DFT), Artificial Neural Networks (ANN), Ion interaction coefficients

Formation of Salt Droplets From Molten $\text{MgCl}_2\text{-NaCl}$ Salt by Helium Bubbling

Dokyu Kang*, Yong Zun Cho, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*kangdk@kaeri.re.kr

Helium bubbling is a method to remove fission products that degrade the performance of molten salt reactor (MSR) from the fuel salt. Helium bubbles can effectively strip noble gases (Xe and Kr) and noble metal particulates from the molten salt and the bubbles containing these fission products are eventually swept into the off-gas system. However, when the bubbles arrive at the free surface, the bubbles will burst, thereby salt droplets will be produced. These salt droplets can cause plugging of functional components in MSR system, a phenomenon that was actually observed in the molten salt reactor experiment (MSRE) conducted by Oak Ridge National Laboratory (ORNL).

In this study, we investigated the formation of salt droplets from molten salt through helium bubbling. Helium bubbling was performed in molten $\text{MgCl}_2\text{-NaCl}$ salt at 600°C with a range of helium flow rate (0.8, 2.4, 4, 6, and 8 L/min at STP) and the salt droplets entrained in the off-gas were analyzed. The size of the salt droplets are mostly distributed between several μm to 30 μm and the mass of the salt droplet is exponential increase with the flow rate (0.46 mg at 0.8 L/min to 171.17 mg at 8 L/min). In addition, the feasibility of removing particulates (MgO and Mo) by the formation of salt droplets was also tested.

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Keywords: Helium bubbling, Molten salt reactor, Salt droplet, Bubble bursting

Purification and Measurement of Oxygen/Oxide Level in MgCl_2 Containing Molten Salt

Dongjun Shin*, Taeho Kim, and Changhwa Lee
Korea Atomic Energy Research Institute (KAERI)
*djshin@kaeri.re.kr

Chloride-based molten salts at high temperatures are an attractive medium for heat transfer and energy storage. MgCl_2 , in particular, possesses a redox potential that is highly similar to that of nuclear fuel salts, thus it is often employed as a surrogate material in nuclear fuel simulations. This research focuses on the development of purification technologies for molten salts containing MgCl_2 , with the aim of enhancing the removal of oxygen/oxide in molten salts. Considering the thermodynamic, physiochemical and electrochemical characteristics of MgCl_2 containing molten salts, various purification technologies were suggested and experimentally verified. To measure the oxygen/oxide level in the salt, elemental analyzer with non-dispersive infrared spectroscopy (NDIR) method was employed. Additionally electrochemical properties of the molten salts before/after purification were monitored using YSZ oxygen sensor. The obtained result were discussed, This study presents new technological approaches for producing high-purity molten salts and aims to contribute to improving the performance of liquid fuels in MSR.

Keywords: MSR (Molten Salt Reactor), Thermodynamics, Salt purification, Selective reduction/oxidation

Catalyst Performance Test System for the Production of Enriched Heavy Water

Hye Jin Cho¹, Nguyen Van Toan¹, Chan Woo Park¹, Dong Woo Lee^{1,2}, Junghwan Park¹, Sang Ho Lim^{1,2},
and Jong-Yun Kim^{1,2,*}

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*k jy@kaeri.re.kr

Heavy water (deuterium oxide, D₂O) is becoming more and more important in some industries, which is a situation that relies on imports, as there are no domestic producers of heavy water. The heavy water is a valuable source for manufacturing components in displays such as organic light emitting diodes (OLEDs), semiconductors, and pharmaceutical applications. However, it is crucial to secure domestic source technology for heavy water production, such as process technology, mass production process equipment, and securing materials. Heavy water can be concentrated in the liquid phase catalytic exchange (LPCE) process, which takes place in a catalytic column where isotopic exchange of hydrogen occurs.

In this work, we design and establish a performance test system liquid phase catalytic exchange (LPCE) process for catalyst reliable analysis of heavy water production capability. The catalytic exchange process was built up with a water chamber reactor monitored by a temperature controller, a main reaction column with a heavy water product ion tank, a cooling column with a liquid product ion tank, and a drying column, respectively. The main column packing was configured for random packing with hydrophobic catalyst and Dixon rings, in which the deuterium gas is injected into the catalyst tower, and water moves from the top of the catalyst tower to the bottom. The final dried deuterium gas from drying column packed with silica gels was directly connected to the GC-MS to analyze the enrichment of the D₂ gas. We installed gas lines, which flow rates are controlled by MFC controllers, a deuterium gas leak detector and alarm, a ventilation fan, and an Ar gas by-pass line to prevent leakage. Additionally, we have developed a novel approach utilizing porous polymer beads as a catalytic support for LPCE, aiming to surpass the constraints of current methodologies. The BET surface area and pore size of the polymer beads are given by N₂ adsorption experiments, demonstrating potential catalysts in LPCE for enhanced deuterium enrichment. This advances can be applied to field processes by achieving optimal catalyst design conditions and modulating the design variables such as laboratory-scale catalyst tower configuration, column height, and catalyst type.

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Keywords: Heavy water, Deuterium oxide, Liquid phase catalytic exchange, Catalyst, Hydrogen isotope

Comparison and Evaluation of Pre-Treatment Methods for Chemical Separation of Sr-90 and Tc-99 Difficult-to-Measure Radionuclides in Decommissioning Waste

Eunmi An, Homyong Lee, and Yeong Su Ha*

Korea Research Institute of Decommissioning (KRID)

*zerowater@krid.re.kr

In the decommissioning process of nuclear power plants, Sr-90 and Tc-99 present in radioactive waste are crucial radionuclides used as fundamental indicators for determining waste treatment methods. However, due to their beta emissions, the chemical separation of these two radionuclides is essential compared to others. Therefore, this study aims to evaluate the suitability of various pre-treatment methods for decommissioning waste materials by conducting characteristic assessments at each chemical separation stage.

The main aim of this study is to find the most optimized pre-treatment method for the analysis of Sr-90 and Tc-99 in decommissioning waste. The comparative evaluation results confirm that the chemical separation procedures for Sr and Tc are very stable in terms of stepwise recovery rates and the removal of interfering radionuclides. However, decommissioning waste materials, which mainly consist of concrete, metals, etc., possess unique properties, and a significant portion may be low-radioactivity waste suitable for on-site disposal. Considering that the chemical behavior and reaction characteristics may vary at each chemical separation stage depending on the matrix properties of the materials, it is considered necessary to apply cascading chemical separation or develop and apply individual chemical separation methods. This should be done by verifying and validating their effectiveness on actual decommissioning waste materials.

Acknowledgements

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Keywords: Radioactivity analysis, Decommissioning waste, Difficult-to-measure, Sr-90, Tc-99

Measurement of Uranium Concentration Using Time-Resolved Laser Fluorescence Spectroscopy (TRLFS) (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle 2024)

Jeongbeen Lee¹, Byungdoo Kang², Seungmo Yeon³, Jisoo Jung³, and Jong-Il Yun^{3,*}

¹Dongguk University

²Kyung Hee University

³Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

In a 2022 survey of private groundwater wells in Korea, 64 out of 4,415 wells (1.4%) were found to have uranium levels exceeding the drinking water quality standard of 30 parts per billion (ppb), as established by the Korean Ministry of Environment. Given that a significant portion of Korea's land consists of granite and metamorphic rocks, monitoring the concentration of uranium in groundwater used for potable purposes is crucial. The current monitoring frequency is once a year, with additional testing conducted over three months if the standard is exceeded. This limited testing frequency is due to the high cost and expertise required for inductively coupled plasma-mass spectrometry (ICP-MS) analysis, which poses challenges to increasing inspection frequency.

Other methods for the quantitative analysis of uranium in aqueous solutions include time-resolved laser-induced fluorescence spectroscopy (TRLFS), liquid scintillation counting (LSC), and radiochemical spectrometry. Among these, TRLFS exhibits excellent sensitivity, with detection limits ranging from 0.02 ppb to 0.05 ppb. It can selectively detect uranyl ions (UO_2^{2+}) based on their fluorescence signal while minimizing interference from other fluorophores through time-resolved techniques. This laser-based analytical technique is particularly useful for rapid determination of uranium prior to sending samples to a laboratory equipped with more complex and expensive instruments such as ICP-MS.

In this study, TRLFS was employed to quantitatively analyze the concentration of uranium in groundwater samples collected from Area A in Daejeon. A standard solution of 10,000 ppm uranium was diluted with 2% high-purity nitric acid to prepare standard solutions with uranium concentrations ranging from 1,000 ppb to 1 ppb. Groundwater samples were also pretreated to a 2% nitric acid concentration, and quantitative analysis was conducted. The uranium concentration of the diluted sample was determined to be 63.63 ppb. Consequently, the uranium concentration in the groundwater was calculated to be 79.54 ppb.

The analysis of groundwater samples collected from Area A in Daejeon using the TRLFS method revealed that the uranium concentration exceeded the regulatory drinking water quality standard of 30 ppb as established by the Ministry of Environment. TRLFS represents a viable alternative to ICP-MS, the conventional method for uranium analysis, and is expected to be particularly useful in situations requiring rapid measurement of uranium.

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Keywords: Uranium, Time-Resolved Laser Fluorescence Spectroscopy (TRLFS), Groundwater

Study on the TCS Effect According to Sample Location in Gamma Nuclide Analysis of Low- and Intermediate Level Radioactive Waste

Seok Han Kim*, Junghyun Park, Seung-Ho Lee, and Yeong-su Ha

Korea Research Institute of Decommissioning (KRID)

*sky@krid.re.kr

In order to process low- and intermediate-level radioactive waste generated during the decommissioning of nuclear power plants, it is necessary to analyze the radioactivity concentrations of gamma nuclides such as Co-58, Co-60, Cs-137, and Ce-144. Gamma rays from continuous decay nuclides such as Co-60 are affected by True Coincidence Summing (TCS) due to the characteristics of the gamma detector. Low- and intermediate-level radioactive wastes are more affected by this effect because they exhibit higher levels of radioactivity than environmental samples, even when diluted. The objective of this study is to investigate ways to reduce the TCS effect based on the positions of the sample and the HPGe detector crystal during gamma nuclide analysis of low- and intermediate-level radioactive waste. If this effect is not considered during calibration, the measurement efficiency decreases, and the radioactivity concentration of a specific nuclide may be overestimated. While there are methods to reduce the TCS effect, such as using a compensation program or a TCS-only calibration source, this study aims to find the optimal configuration to reduce the TCS effect by altering the geometry of the HPGe crystal and sample, and to explore this calibration method for more accurate gamma nuclide analysis. Using various types of certified standard sources provided by KRISS, we aim to investigate the effects of HPGe detector crystal shape, efficiency according to the distance from the sample, True Coincidence Summing correction, etc., and utilize these findings for radioactive waste gamma nuclide analysis.

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Keywords: HPGe, True coincidence summing, Radioactive waste, Calibration

Characterization of Uranyl Nitrate Solution With Laboratory-Based EXAFS (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle 2024)

Minsik Jeon¹, Seokjoo Yoon², and Jong-Il Yun^{2,*}

¹Sejong University

²Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

The management of radioactive waste generated during the nuclear fuel cycle has become a significant concern in recent years, as nuclear power plants approach decommissioning. The necessity for the design of appropriate waste forms and storage solutions for long-term waste disposal has increased concurrently.

If spent nuclear fuel, low-level, and intermediate-level radioactive waste are disposed underground, a comprehensive understanding of the chemistry and structure of key radioactive elements is essential regarding the nuclear fuel cycle. These include fuel reprocessing, waste management, reactor safety, and non-proliferation efforts. Nevertheless, the local structure of many actinides remains unknown, making their analysis challenging. Applying X-ray absorption fine structure (XAFS) is recommended to address this issue. By analyzing extended X-ray absorption fine structure (EXAFS) and X-ray absorption near edge structure (XANES), it is possible to determine the local structure surrounding the atom.

Laboratory-based XAS equipment was introduced for expedient and straightforward discrimination. This method can measure EXAFS and XANES within the energy range of 5-18 keV, offering a more efficient alternative to traditional XAFS techniques. Uranyl nitrate solution and Zirconium standard foil were employed to validate the laboratory-based XAFS within the specified energy range. The EXAFS spectra of the measured compounds agree with the theoretical data. The results demonstrate a covalent bond length of 1.74 Å (U = O) for oxygen covalently bound to uranium and 2.35 Å (U...O) for oxygen in water electrochemically bound to uranium, which is in good agreement with other values reported in the literature. Consequently, this type of analysis will be of significant value in back-end nuclear fuel cycle applications, where understanding the local structure of actinides and major radioactive elements is essential.

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Keywords: X-ray absorption spectroscopy, Local structure analysis, EXAFS, XANES

A Study on the Analysis Results of Spent Activated Carbon From Nuclear Power Plants

Ji Eun Park*, Kahee Jeong, Sooyeon Lim, Yujeong Shin, and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*Jepark12@khnp.co.kr

In nuclear power plants, activated carbon is used in air purification systems to remove gaseous radioactive nuclides generated during operation. For the disposal of spent activated carbon, an evaluation of radioactive nuclides is required, with the primary focus on carbon-14 and tritium.

This study assesses the radioactivity of carbon-14 and tritium in spent activated carbon from nuclear power plants. Common methods for analyzing the radioactivity of tritium and carbon-14 include wet extraction or pyrolysis, depending on the sample type.

Considering the porous nature of spent activated carbon, this study applied the pyrolysis method. By heating the samples above 800°C using a pyrolysis device, carbon-14 was oxidized to carbon dioxide, and tritium was vaporized as HTO. The gases produced were absorbed into respective collection solutions, and the radioactivity was measured using a liquid scintillation counter after mixing the solutions with an appropriate scintillation cocktail. This study explores the characteristics of spent activated carbon, the selection of the appropriate analytical methods based on these characteristics, and a discussion of the results obtained from the analysis.

Keywords: Spent activated carbon, Tritium, Carbon-14, Pyrolysis

Electrochemical and Physical Behaviors of ZrCl_4 in Molten NaCl-KCl Salt

Kyeongtae Park and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Molten salt reactor (MSR), one of the 4th generation reactors applies molten salt media which works both as fuel and heat exchanger. It is also good system for preventing severe accident by solidification of salt at loss of coolant situation. However, by the aggressively corrosive properties of salt with structural material, many of research are being conducted to ensure the material compatibility with salt. Representative factor affecting on material corrosion in varied of moisture, actinide ratio, redox condition, etc. One of the recently introduced factor for MSR system design is the species and ratio of fission products produced on MSR operation. These additives would affect the redox condition of salt and furthermore on structural material compatibility with salt. In this study, one of the candidates on MSR fission products, zirconium tetrachloride (ZrCl_4) was investigated on the aspect of basic chemical reactions with molten NaCl-KCl salt and electrochemical behavior in the salt.

Whole process of experiments was conducted in the well controlled glove box with less than 0.1 ppm of oxygen and moisture. First, solution stability was tested for confirming the ZrCl_4 would rest on salt for targeted or long-term operation of MSR. Various concentration of ZrCl_4 was inserted to molten NaCl-KCl eutectic salt in 700°C and stayed in liquid. Sample salts were collected on various time after inserting the ZrCl_4 with cold quartz rod immersion, followed by concentration analysis with ICP-OES. Electrochemical properties were collected with three electrode system with tungsten working/counter electrode and reference electrode for silver wire with $\text{NaCl-KCl-1wt\% AgCl}$ salt in quartz membrane. Cyclic voltammetry was conducted with various ranges in 0.2 V to -2.0 V of boundary and could detect each oxidation and reduction peaks of zirconium. Chronoamperometry was conducted with the first peak of oxidation, estimated to reduce ZrCl_4 to ZrCl_2 and after salt was analyzed for checking disproportionation would occur with ZrCl_4 in molten salt.

ZrCl_4 showed very complex redox behavior in molten NaCl-KCl salt from this study. So, much more various confirmations on zirconium behavior should be conducted to clearly clarify how it affects on the salt condition in MSR. For further study, corrosion behavior of structural material candidates of MSR with ZrCl_4 should be considered to ensure the safe and stable long-term operation of MSR.

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Keywords: Molten salt reactor, Fission products, Impurity, Electrochemistry, Corrosion

Processing of Individual Particle Data for True Coincidence Summing Evaluation in Gamma-ray Detection Using MCNP 6.3

Kyunghun Jung*, Simon Oh, Woojin Jo, and Jung Bo Yoo

Korea Atomic Energy Research Institute (KAERI)

*kyunghunjung@kaeri.re.kr

The safe disposal of radioactive waste requires analyzing the activity of alpha, beta, and gamma-emitting nuclides in waste drums, as regulated by the Nuclear Safety Commission. Among various methods, the HPGe gamma-ray detector is the most widely used instrument for measuring gamma nuclide activity. The HPGe detector collects gamma rays emitted by nuclides in the sample and produces an energy spectrum, enabling simultaneous analysis of multiple nuclides. To accurately determine these activities, the detector's efficiency must be known, which is typically calibrated using a certified reference material (CRM) with known activity and a similar composition to the sample. However, if the CRM contains nuclides like Co-60 or Y-88, which can cause true coincidence summing (TCS), it becomes challenging to evaluate the sample's activity accurately. To correct for this, computer simulations are often employed to account for TCS effects in measurement efficiency. At KAERI, we are investigating methods to correct for coincidence summing using the Monte Carlo N-Particle (MCNP) code. MCNP simulates particle interactions under specific conditions and is commonly used to evaluate the efficiency of gamma detectors. It runs multiple simulations of particle transport, processing the results statistically to generate a tally. However, MCNP can only simulate the transport of individual particles and cannot directly model the TCS caused by interactions between multiple particles. By using MCNP's PTRAC function, which records detailed transport information for individual particles without statistical processing, we can assess coincidence summing by analyzing the timing of particle events. In MCNP versions up to 6.2, PTRAC results were saved in ASCII format, making it difficult to parse and handle large datasets. With the release of MCNP version 6.3, PTRAC output has shifted to the HDF5 format, which is optimized for large-scale data storage and allows faster input/output and easier data processing compared to the older ASCII format. In this presentation, we will share the initial findings of our study on using MCNP for TCS corrections. We extracted data from PTRAC files in HDF5 format and compared the statistical results with those obtained from traditional tally outputs to assess consistency.

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Keywords: MCNP, True coincidence summing, HPGe gamma detector, Detector efficiency

Evaluation of the Representativeness and Accuracy of Cold-Finger Method for Molten Salt

Dongmin Kim, Jihye Park, Hyeongbin Kim, Wonseok Yang, and Sungeol Choi*

Seoul National University (SNU)

*choisys7@snu.ac.kr

Molten Salt Reactors (MSRs) are a type of advanced reactor where both coolant and fuel are molten salts, and remain mixed together in the reactor core. In MSRs, nuclear fuel is directly dissolved into the coolant, resulting in fission products also remaining mixed into the coolant. These fission products need to be monitored for safety and efficiency. Therefore, real-time monitoring of fission products within the molten salts is essential, and Laser Induced Breakdown Spectroscopy (LIBS) is one of the strong tools for measurement techniques in molten salt environments. LIBS is an analytic method that uses a laser to create plasma for the sample, allowing for rapid and precise elemental analysis. Since LIBS allows for quick, in-situ measurements with minimal sample preparation in complex environments, it is a highly suitable choice. To accurately analyze the composition, the best approach is to direct the laser just above the liquid surface, however, when LIBS is directly applied to liquid samples, it suffers from high standard deviation (STDEV) due to non-uniform plasma formation, leading to lower accuracy. Therefore, cold finger sampling methods were generally performed for molten salt analysis.

This study aims to investigate how to prepare solid samples that most accurately replicate the behavior of liquid molten salts in LIBS analysis. Additionally, as the matrix effect, a well-known limitation of LIBS, can vary with each sample preparation methods, the study examined how matrix effects influence the analysis for each solid sampling techniques to improve the reliability of solid sample LIBS. Various geometries of tungsten (rod, square plate, and ladle) were used for solid sampling to investigate the effects of different solid salt formations. SrCl_2 , a high decay heat fission product, concentration was measured in varying concentrations from 0 to 10,000 mg/kg within the LiCl-KCl salt. Liquid sample LIBS was performed inside the glovebox to achieve an Argon atmosphere environment ($[\text{O}_2]$, $[\text{H}_2\text{O}] < 3$ ppm); solid sample LIBS was performed outside the glovebox after sampling. The most effective solid sampling method was identified and validated against Inductively Coupled Plasma Optical Emission Spectroscopy (ICP-OES). This approach provides calibration curves and detection limits (LOD) that most accurately describe those of liquid sample analysis, thereby enhancing the potential for practical and accurate real-time monitoring.

Acknowledgements

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Keywords: Laser-Induced Breakdown Spectroscopy (LIBS), Cold finger sampling, Molten Salt Reactor (MSR), Solid LIBS

A Study of Gamma MDA Reduction in Accordance With Cobalt Removal in Radioactive Spent Resin

Sooyeon Lim*, Kahee Jeong, Ji-eun Park, and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sooyeonlim@khnp.co.kr

During the operation of nuclear power plants, high-level radioactive waste, such as spent resin and sludge, is generated. The majority of the radioactivity in spent resin comes from Co-60, a significant corrosion product formed when Co-59, found in materials like stainless steel and Inconel, reacts with neutrons during reactor operation, leading to the generation of Co-60 in the reactor coolant system. Since spent resin acts as a filter to purify the coolant, it predominantly captures Co-60.

To safely dispose of radioactive spent resin at KORAD (Korea Radioactive Waste Agency), a comprehensive radionuclide inventory must be conducted. This includes the radiochemical analysis of 14 essential nuclides, covering alpha, beta, and gamma emitters. However, because Co-60 emits gamma rays at energies of 1,173 keV and 1,332 keV, a Compton continuum—caused by scattered gamma rays—forms in the region below 1,173 keV. This continuum often leads to the overestimation of the Minimum Detectable Activity (MDA) for nuclides emitting gamma rays just below this energy.

To address this issue, this study focused on minimizing the impact of Co-60 by chemically removing cobalt from the sample. The reduction in Co-60 radioactivity was confirmed, and the changes in the MDA of gamma-emitting nuclides below 1 MeV were observed. A chemical separation method, developed by the research team, was used to selectively remove cobalt while preserving other gamma-emitting nuclides. This method involved extraction chromatography using an anion exchange resin, without the need for hydrofluoric acid.

Keywords: Radioactive waste, Radiochemical analysis, Metallic sample, Radioisotope inventory

Modeling of HPGe System Based on Monte Carlo Simulation

Kahee Jeong^{1,*}, Sooyeon Lim¹, Ji Eun Park¹, Young-Su Kim², and Wook Sohn¹

¹Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

²Korea Atomic Energy Research Institute (KAERI)

*kaheejeong@khnp.co.kr

Low and intermediate level radioactive waste samples from nuclear power plants inevitably contain high-energy nuclides with high radioactivity concentrations, such as Co-60. The background signals from the Compton continuum of these nuclides makes it difficult to analyze the peaks of trace nuclides. This increases the minimum detectable activity (MDA) and may result in reporting high MDA values even when trace nuclides are present in low quantities. Instead of increasing the amount of analysis sample or the measurement time to reduce MDA, we aim to develop a technology to reduce MDA of target trace nuclides by removing the Compton continuum effect of interfering nuclides in the gamma spectrum. We introduce a Monte Carlo simulation-based model for HPGe systems, which has been validated to show no statistically significant difference from actual measurement results.

Evaluation of Gross Alpha Radioactivity by Spillover

Kahee Jeong*, Sooyeon Lim, Ji Eun Park, and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kaheejeong@khnp.co.kr

The gas proportional counter is a measurement device used for gross alpha and gross beta analysis, with the advantage of distinguishing pulse signals from alpha and beta particles. This pulse discrimination is achieved by adjusting the detector's operating pressure and setting the electronic discriminator to reject pulses below a certain threshold. In the gross alpha/beta simultaneous measurement mode, since the equipment operates at beta mode voltage, the overlap rates of alpha and beta must be considered, with the manufacturer-specified ratio generally being less than 0.1%. However, this ratio does not guarantee perfect separation of alpha and beta signals.

In this study, the impact of spillover was evaluated using alpha only and alpha-beta simultaneous measurement modes for samples with high beta radioactivity. The results suggest using the alpha-only measurement mode for gross alpha analysis.

Preliminary Test of Gamma and Gross Alpha Specific Radioactive Evaluation for Low-Radioactivity Waste Resin Disposal

Kahee Jeong*, Sooyeon Lim, Ji Eun Park, and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kaheejeong@khnp.co.kr

A disposal plan for spent resin, subject to first-time disposal using the 860 L polymer concrete high-integrity container (PC-HIC), has been established following the completion of the KORAD's second construction project.

This study was conducted to generate basic data for low-radioactivity waste resin samples by determining the sample quantity for analysis and performing gross alpha radioactivity analysis. The aim was to verify the gamma nuclide analysis results of the dissolved samples after pretreatment and to assess the necessity of individual alpha nuclide analysis through gross alpha radioactivity evaluation.

Effects of Radiolysis Products on pH Changes in Iodine and Organic Acid Solutions

Jei-Won Yeon

Korea Atomic Energy Research Institute (KAERI)

yeonysy@kaeri.re.kr

Water molecules are subject to radiolysis when exposed to ionizing radiation. Although the acidic and basic species generated by this process are typically produced in nearly equal amounts, resulting in no net change in pH for pure water, this scenario rarely occurs in practice. In most cases, the radiolysis products interact with other chemical species present in the aqueous solution, leading to observable pH changes after irradiation. Since pH is a crucial factor influencing various chemical reactions, understanding these changes is important.

This study investigates pH variations in several aqueous solutions under gamma-ray irradiation and provides a chemical interpretation of the experimental results. In alkaline solutions, the air (N_2+O_2) lowers the pH due to the presence of radiolysis products like NO_2 . In acidic solutions containing iodide (I^-) ions, pH increases as OH radicals or H_2O_2 oxidize I^- to I_2 , generating OH^- ions. On the other hand, in alkaline solutions with iodide ions, the predominant reaction is the oxidation of H_2O_2 , leading to a decrease in pH. For aqueous solutions of organic acids, such as acetic acid (CH_3COOH) or formic acid ($HCOOH$), the pH increases because the organic acids are decomposed by oxidizing radiolysis products of water like OH radicals.

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Keywords: Radiolysis products, Air, Iodide, Acetic acid, Formic acid

Electrochemical Analysis of Uranium Intermetallic Compounds in Molten LiCl-KCl Salts (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle 2024)

Gun-Woo Yoon¹, Kwangseo Kim², Jae-An Lee³, Han-Min Lee⁴, Jun Woo Park⁵, and Jong-Il Yun^{5,*}

¹Hanyang University

²Seoul National University (SNU)

³Sejong University

⁴Kyung Hee University

⁵Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

Material corrosion at high temperatures is one of the key challenges in molten salt reactors (MSR) and pyroprocessing. Chromium (Cr) and iron (Fe), the main components of structural materials, are particularly important in terms of corrosion due to their lowest and second-lowest oxidation potential values, respectively, under molten salt conditions. As corrosion progresses, the concentrations of Cr and Fe increase, potentially affecting the systems through interactions between these corrosion products and uranium (U), the major element in MSR and pyroprocessing. Therefore, understanding the U-Cr and U-Fe reactions is crucial. To confirm the formation of intermetallic compounds and to investigate the redox properties of FeCl_2 and CrCl_2 at high temperatures under LiCl-KCl-UCl_3 molten salt conditions, a series of cyclic voltammetry (CV), a powerful and widely used electrochemical technique, were conducted.

A three-electrode system was used. A 1 mm silver (Ag) wire was dipped into the LiCl-KCl melt with 1wt% AgCl in a thin-end pyrex tube as the Ag/AgCl reference electrode (RE). Cylindrical tungsten (W) electrodes were used as a working electrode (WE) and counter electrode (CE). Full scan and regional scan CVs of LiCl-KCl-UCl_3 , $\text{LiCl-KCl-UCl}_3\text{-CrCl}_2$, and $\text{LiCl-KCl-UCl}_3\text{-FeCl}_2$ were conducted at a scan rate of 200 mV/s, repeat cycle 3 times at single cycle and 500°C. All the experiments were conducted in a glove box. Additional measurements were conducted by adding FeCl_2 and CrCl_2 to verify whether the observed redox peaks were due to intermetallic compound formation.

Chemical window of LiCl-KCl was confirmed in the range of -2.42 V ~ 1.35 V, and redox peaks of U(III)/U(0) , U(IV)/U(III) , Cr(II)/Cr(0) , Cr(III)/Cr(II) and Fe(II)/Fe(0) were observed. As in the U-Cr intermetallic phase diagram, no U-Cr intermetallic compound was present in the 500°C $\text{LiCl-KCl-UCl}_3\text{-CrCl}_2$, and as in the U-Fe intermetallic phase diagram, two U-Fe intermetallic compounds were formed in the 500°C $\text{LiCl-KCl-UCl}_3\text{-FeCl}_2$, confirming that two pairs of redox peaks of U-Fe were present in -1.4 V ~ 0.5 V.

It was confirmed that no intermetallic compound of U-Cr exists, and two intermetallic compounds of U-Fe exist at 500°C. This experiment shows electrochemical windows, provides the basic knowledge for corrosion detection in MSR pipes and pyroprocessing, and confirms that energy consumption can be optimized when U is selectively recovered.

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Keywords: Cyclic Voltammetry (CV), Redox peak, Intermetallic compound, Electrochemical window, Molten Salt Reactor (MSR), Pyroprocessing

Synthesis and Analysis of C-S-H Colloids Using Multi-Angle Dynamic Light Scattering and Electrophoretic Light Scattering (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle 2024)

Gyutaek Kang¹, Eunbi Kang², Jihu Lee², Chongmyung Jin³, Hyeonjin Eun⁴, Seungwoo Lee⁴, and Jong-Il Yun^{4,*}

¹Sejong University

²Dongguk University

³Seoul National University (SNU)

⁴Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

When assessing the safety of radioactive waste repositories, the possibility of radionuclide migration through engineered barriers should be considered. In the Gyeongju Low and Intermediate-Level Radioactive Waste (LILW) repository, calcium silicate hydrate (C-S-H) colloids are formed from the hydration reactions of cement, the primary component of concrete. C-S-H colloids can bind with radionuclides, potentially accelerating their migration. Therefore, assessing the stability of C-S-H colloids is essential for ensuring the long-term safety of radioactive waste disposal sites.

C-S-H was synthesized by mixing equal volumes of 0.05 M Na_2SiO_3 and 0.05 M $\text{Ca}(\text{NO}_3)_2$, followed by vacuum filtration and vacuum drying. In addition, artificial rainwater and concrete pore water were prepared, into which 1 g/L of the synthesized C-S-H was suspended. The stock solutions of rainwater and concrete pore water were compared with their respective C-S-H suspensions. Multi-Angle Dynamic Light Scattering (MADLS[®]) was used to determine the hydrodynamic diameter and the concentration of the colloids present in solution, whereas electrophoretic light scattering (ELS) was used to determine their zeta potential. MADLS[®] revealed the particle concentrations to be 2.4×10^4 particles/mL near 430 nm for rainwater, 1.6×10^7 particles/mL near 430 nm for C-S-H rainwater suspension, 1.1×10^5 particles/mL near 350 nm for concrete pore water, and 2.7×10^7 particles/mL near 460-530 nm for C-S-H concrete pore water suspension. Particle concentrations within 430-530 nm size range were significantly higher in C-S-H suspended solutions, indicating that C-S-H particles remain chemically stable in rainwater and concrete pore water without dissolution. The zeta potentials of the C-S-H rainwater suspension and C-S-H concrete pore water suspension were -19 mV and -23 mV, respectively. The magnitude of the zeta potentials implies that aggregation of C-S-H colloids, within rainwater and concrete pore water, are hindered by electrostatic repulsion. The pH of rainwater, C-S-H rainwater suspension, concrete pore water and C-S-H concrete pore water suspension were 6.5, 10.2, 11.4, and 10.8, respectively. The increase in pH of rainwater on C-S-H addition, as well as the similar zeta potentials of the two C-S-H suspensions, suggest that rainwater begins to yield concrete-pore-water-like properties upon contact with C-S-H.

The results of this study allowed the evaluation of C-S-H colloid stability, which is expected to be valuable information when assessing the long-term safety of the Gyeongju LILW repository.

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Keywords: C-S-H colloids, Radionuclide migration, Stability, Particle concentration, Zeta potential

Estimating the Electroneutrality of Fission Products in Molten Chloride Fast Reactor

Seokjoo Yoon, Jun Woo Park, Hyeonjin Eun, and Jong-Il Yun*

Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

Molten salt reactors will exhibit a more complicated situation in terms of the behavior of fission products to achieve electroneutrality. Compared to conventional solid-fuel reactors where the interaction of the fission products outside of the fuel matrix is limited, the interaction of the fission products in molten salt with the environment will become critical, especially regarding electroneutrality. For instance, if the electroneutrality of the system cannot be achieved after one fissile isotope undergoes fission, then the system will further oxidize the system, mostly the structural material. The escape of the noble gas will also play a crucial role regarding electroneutrality since the decay of the noble gas produces fission products that become cationic species in the molten salt system. Therefore, estimating the electroneutrality in the molten salt reactor should be performed.

This research investigates the electroneutrality of the fission products in the molten chloride fast reactor. The electroneutrality of the fission product from the fast fission of U-235, U-238, and Pu-239 that can be utilized in chloride-based molten salt reactors is analyzed. Different scenarios are assumed regarding the escape of the noble gas in the system. Compared to the U-235 where electroneutrality could be achieved without the oxidation of the structural material, the fission of U-238 and Pu-239 will require further oxidation of the structural material to reach the electroneutrality. The removal of noble gas shows a negative effect regarding electroneutrality, as the decay of the noble gas elements leads to the formation of cationic species which plays an important role in achieving electroneutrality. These results imply that reactors utilizing the fast fission of U-238 and Pu-239 can induce further oxidation of structural material apart from the corrosion in the molten chloride system, and measures to prevent oxidation from fission reaction should be considered in the molten salt reactors.

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Keywords: Electroneutrality, Fission products, Molten chloride fast reactor

Solvent Effects on Electrochemical Reduction of U(VI) to U(IV) in Nitric Acid and Perchloric Acid Solutions

Sehwa Bae¹, Seungwoo Lee², Hyeonjin Eun², Jun Woo Park², and Jong-Il Yun^{2,*}

¹Chungnam National University (CNU)

²Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

Understanding the chemical behavior of tetravalent actinides such as U(IV), Np(IV), and Pu(IV) under reducing geochemical conditions is crucial for the safe disposal of spent nuclear fuel. This study investigates the electrochemical reduction of U(VI) to U(IV) in nitric acid and perchloric acid solutions.

Two U(VI) solutions were prepared by dissolving $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ in 5 M HNO_3 with 100 mM NH_2OH , and in 1 M HClO_4 . The concentration of U(VI) in each solution were 50 mM and 36 mM, respectively, as determined by ICP-MS. 50 mL of each of the solutions were reduced by potentiostatic electrolysis using the same three-electrode system. All mentioned potential values are against a Ag/AgCl (3 M KCl) reference electrode.

In the HNO_3 solution, the cyclic voltammograms revealed a two-step electron transfer process for the reduction of U(VI), showing distinct reduction peaks at 0.20 V (U(VI) to U(V)) and -0.15 V (U(V) to U(IV)). In contrast, in the HClO_4 solution, a single reduction peak was observed at -0.25 V (U(VI) to U(IV)), corresponding to a simultaneous two-electron transfer from U(VI) to U(IV). Based on these results, electrochemical reductions were performed at -0.20 V for the HNO_3 solution and -0.30 V for the HClO_4 solution. The reduction of U(VI) to U(IV) was quantified using UV-Vis absorption spectroscopy.

The UV-Vis absorbance of U(VI) in the HNO_3 solution decreased slightly and remained constant after 24 hours, showing a maximum reduction yield of 27%. Conversely, in the HClO_4 solution, the U(VI) absorbance was no longer visible, and the reduction yield was near 100%. As the same three-electrode system was used, the significant change in the reduction percentage was due to the solvent effect.

The difference between the HNO_3 and the HClO_4 solutions can be understood by constructing Pourbaix diagrams. The Pourbaix diagrams revealed that $\text{UO}_2(\text{N}_3)_4^{2-}$, a hexavalent uranyl complex, was the dominant uranium species in the HNO_3 solution. In contrast, U^{4+} was the predominant species in the HClO_4 solution.

The results demonstrate that the redox couple $\text{NO}_3^-/\text{N}_3^-$ constraints the reduction of U(VI) to U(IV) in HNO_3 . Despite the addition of NH_2OH to prevent the accumulation of HNO_2 in the HNO_3 solution, HClO_4 was found to be a more suitable solvent for the reduction of U(VI) to U(IV).

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Keywords: Electrochemical reduction, U(VI), U(IV), Solvent effects

New Methodology for Describing Oxidizing Condition of Used Chloride-based MSR Salt

Ju Ho Lee* and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*jhlee835@kaeri.re.kr

The development of design and operational technologies for molten salt reactors (MSRs) requires information about the chemical forms of fission products under specific conditions and their distribution among phases. Conventional methodology employs CALPHAD-based thermodynamic codes coupled with thermodynamic databases such as MSTDB. These codes require fission product inventories obtained from neutronic codes like OpenMC to identify the available chemical species and their distribution within existing phases. However, these calculations must be carefully evaluated by comparing them with real-world physical phenomena.

One such phenomenon in MSRs is the oxidation of the molten salt during reactor operation. This study evaluates the applicability of the conventional methodology using FactSage 8.3 coupled with MSTDB v3.1 to describe the oxidizing conditions within a 100 MW NaCl-KCl- UCl_3 based MSR operating for 30 years. Initially, the total inventory of all fission products was calculated using OpenMC, accounting for depletion during the neutronic calculations. Subsequently, 44 element species from uranium to dysprosium were selected and input into the equilibrium module of FactSage along with their masses. The equilibrium calculations revealed that zirconium metal exists in the solid phase, implying a reducing environment in the salt. To correct this discrepancy, a new methodology was introduced. It re-evaluated the inventory, accounting for the partial release of fission products with noble gas precursors into the gas phase. Additionally, MSTDB was revised by including model parameters for seven binary systems previously neglected: ZrCl_4 -KCl, SmCl_3 -KCl/NaCl, YCl_3 -KCl/NaCl, and PrCl_3 -KCl/NaCl. The calculations based on this new methodology demonstrated an oxidizing environment with no zirconium metal in the solid phase and a significantly increased ratio of UCl_4 to $\text{UCl}_3 = 1.2 \times 10^{-4}$.

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Characterization of Soil-derived Colloids in the Aquatic Environment: Colloid Stability Depending on Clay Minerals and Solution Chemistry

Kyeong Min Lee, Youngbeom Kim, Su Chang Kim, Hye-Ryun Cho, Sang-Ho Lee, and Hee-Kyung Kim*

Korea Atomic Energy Research Institute (KAERI)

*hkim11@kaeri.re.kr

Colloidal particles can play a role as carriers of radionuclides (RNs) in aquatic systems and thus accelerate migration of RNs. To establish acceptance criteria for particulate wastes such as soils and glass fibers for the repository systems of low-to intermediate-level radioactive wastes (LIRW), it is required to understand the properties of particulate waste-derived colloids and their interactions with RNs and with engineering/natural barriers. This study aimed to characterize soil-derived colloids in the aspects of colloid formation mechanism and stability under various conditions, and evaluate their potential role in RN transport within aquatic environments of disposal systems. The research focused on three clay minerals rich in soils in South Korea: Kga-1b (kaolinite-dominant), ISCz-1 (70% illite, 30% smectite, $d < 0.15$ μm), and Y.I.L (illite-dominant, $d < 0.15$ μm). B.WRK (70% smectite, 30% others) was also examined as a reference material. Mineral compositions of the source samples were analyzed by using XRD. Colloids were extracted by shaking the source samples (10 g/L) in distilled water, synthetic concrete pore water (SynCPW, $I_m = 3.2$ mM, pH 11.5), and 5.0 mM NaOH solution (pH 11.6) at 120 rpm for 24 hr. After 3-days of settlements, dispersed colloids in the supernatants were fractionated depending on the size by sequential centrifugations with increasing g forces. The stability of the fractionated colloid samples was monitored over 30-60 days; undisturbed samples were periodically aliquoted from the upper part, and subjected to the analysis of concentrations, size, and zeta-potential using ICP-AES, dynamic light scattering (DLS), and nanoparticle tracking analysis. Results showed distinct stability patterns depending on the samples influenced by mineralogy and solution chemistry. In general, Kga-1b clay produced too low amount of colloids to be analyzed by DLS, while other clays produced considerable amounts of colloids. In particular, colloids generated in 5.0 mM NaOH control medium were remarkably stable with a zeta-potentials below -40 mV and size distributions in the range of 150-300 nm. Notably, all colloid types became unstable in SynCPW, which is ascribed to the charge neutralization effect by the calcium ions (1.3 mM) in SynCPW. The results highlight the significant impact of solution composition on colloid stability. Detailed experimental results will be presented and the behavior of soil-derived colloids under conditions relevant to LIRW disposal systems will be discussed.

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Keywords: Colloid stability, Clay mineral, Zeta-potential, Dynamic light scattering, Nanoparticle tracking analysis, ICP-AES

Measurement of Rheological Characteristics of Various Molten Salt

Hyeoneui Hong, Taehoon Park, Wonseok Yang, Jihun Kim, and Sungyeol Choi*

Seoul National University (SNU)

*choisys7@snu.ac.kr

Molten salt reactors (MSRs) offer high thermal efficiency due to their high operating temperatures (500–800°C) resulting from the use of molten salts. Additionally, the risk of accidents is reduced because the molten salt not only serves as the coolant, but also as a carrier medium for the nuclear fuel which is dissolved directly in the molten salt coolant. However, understanding the rheological properties of the molten salts in MSRs is crucial, as these properties change depending on the components of molten salts. Specifically, the density and viscosity of molten salts are critical in controlling reactivity and are key factors to consider when selecting materials for reactor design. These properties play an important role not only during the normal operation of MSRs but also in AOO (anticipated operational occurrence) and DBE (design-basis event) scenarios.

In this study, we measured the temperature-dependent density and viscosity of KNO_3 , LiCl-KCl eutectic salt, and NaCl-MgCl_2 eutectic salt using a rheometer (MCR 302e, Anton Paar) under an argon atmosphere. We chose the rheometer for its capability to measure both density and viscosity simultaneously over a wide temperature range (25–1,000°C). This equipment measures the density using the Archimedes method and calculates the viscosity by varying the shear rate applied to the bob and measuring the torque exerted on it. The temperature range for each salt was set to reflect the temperature at which it is primarily used. There is no standard reference material for molten salts, making it challenging to verify the precision and accuracy of our measurement equipment and procedures. Measured values were verified by comparing to KNO_3 (345–460°C), which has been previously studied, we found that the errors were not significant. This allowed us to validate the precision and accuracy of our equipment. Subsequently, we measured the properties of widely used molten salts such as LiCl-KCl eutectic (400–700°C) and NaCl-MgCl_2 eutectic (500–800°C).

In conclusion, the rheological properties of various salts were measured and verified with experimental data. Other types of molten salt mixtures, such as NaCl-KCl , KCl-MgCl_2 will be investigated in the future.

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Keywords: Temperature-dependent, Physical properties, Molten salt, Density and viscosity

Evaluation of True Coincidence Summing Effects With Varying Detection Geometry in Gamma-Spectroscopy

Junhyuck Kim*, Hye-Ryun Cho, and Jae Il Park

Korea Atomic Energy Research Institute (KAERI)

*jhkim98765@kaeri.re.kr

Accurate measurement of gamma-ray emitting isotopes in low-level samples, such as soil samples from decommissioning and environmental remediation of nuclear facilities, is important for radiation safety and environmental protection. However, various factors can affect the measurement accuracy of these low-level gamma-samples such as True Coincidence Summing (TCS) effect, which is correlated with gamma-peak detection efficiency. This study aims to evaluate the influence of TCS effect on various detection geometries and to prioritize considerations for practical analysis.

This study examines the impact of TCS effect based on different detection geometries, such as sample shape and the sample-detector distance, and using a High-Purity Germanium (HPGe) detector with relative efficiency of 30%, which is commonly used in gamma-spectroscopy. Monte Carlo simulations were used to derive the detection efficiency curve, and the software for calculating TCS correction factors was utilized. The geometric conditions with Marinelli beaker and cylindrical bottle commonly used for analyzing low-level environmental samples were simulated and compared. The correction factor for 1,332.5 keV Co-60 gamma-ray peak with a 450 mL Marinelli beaker was calculated to be 0.92 when placed in front of the HPGe detector window. In other case, the correction factor for the same gamma-ray peak with a 40 mL bottle sample placed 10 cm in front of the HPGe detector, to reduce TCS effect, was determined to be 0.99. Therefore, optimal measurement conditions should be considered with the shape of sample container and the sample-detector distance.

Keywords: True Coincidence Summing Effect (TCS), Gamma-ray spectroscopy, HPGe detector, Low-level radioactive sample, Minimum Detectable Activity (MDA)

Development and Implementation of Laboratory Information Management System (LIMS) for Enhancing Analysis Reliability in KHNP CRI

Sooyeon Lim*, Kahee Jeong, Ji-eun Park, and Wook Sohn

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sooyeonlim@khnp.co.kr

In order to deliver and dispose of low and intermediate-level radioactive waste, it is essential to determine the quantities of radioactivity in the waste. For this purpose, 31 radionuclides, including 14 mandatory radionuclides, must be analyzed using radiochemical methods. Due to the different radiological and chemical characteristics of each radionuclide, the analysis procedure is highly intricate and complex.

Ensuring the reliability of the data generated during this analysis process is critical. There has been a case that radionuclide analysis errors occurred, and as a result, regulatory authorities have demanded that analysts guarantee the reliability of their analyses. Thus, establishing data reliability in radionuclide analysis has become an essential part in the whole process of analysis.

In response, our research team has started developing an advanced radionuclide analysis method, alongside the development of a radionuclide analysis data management system to improve data reliability. This system is based on the currently developed radionuclide analysis workflow. It directly handles data generated from measurement instruments, thereby building a systematic framework.

The radionuclide analysis workflow during radiochemical analysis is generally divided into pre-treatment, chemical separation, and measurement stages. The system has been developed to allow data to be entered and processed in accordance with the flow of sample reception and actual experimentation. Additionally, a reviewer verification and approval step has been added, enabling internal review to occur simultaneously with the experiment. So that system is designed to reduce human error and minimize potential errors during data processing.

Keywords: Radioactive waste, Radiochemical analysis, LIMS, Data reliability

~~Development of an Automated Platform for Reliable Data Collection in Molten-Salt Experiments~~

~~Saehyun Choi, Jihun Kim, Wonseok Yang, Taehoon Park, Richard I. Foster, and Sungyeol Choi*~~

~~Seoul National University (SNU)~~

~~*choisys7@snu.ac.kr~~

Withdrawal

Corrosion Behavior of Inconel 625 in NaCl-KCl-MgCl₂ Molten Salt: Effects of Heat Treatment and Electrochemical Dehydration

Joonha Park, Kyeongtae Park, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

A molten salt reactor (MSR) is one of Generation-IV reactors, which uses molten salt as nuclear fuel and coolant. First, since it operates at a high temperature as salt melts, it has high thermal efficiency. Secondly, if an accident like the loss of coolant accident (LOCA) happens, radioactive material will not spread out because salt will solidify due to its thermophysical properties. Lastly, MSR is mainly considered a fast neutron spectrum, allowing the combustion of transuranic (TRU), which is high-toxic and long-term half-lives. So, it has various advantages in efficiency, safety, and management of radioactive waste.

Despite its advantages, the commercialization of MSR has been delayed due to concerns about corrosion of structural material in molten chloride salt. The factors that affect corrosion of the material are moisture compound, redox potential change in salt, and irradiation. However, since the study of chloride salt has not progressed much compared to fluoride salt, it is essential to study the moisture effect on chloride salt.

In this study, the corrosion behavior by moisture level in molten chloride salt was investigated. Corrosion experiments were conducted with Inconel 625 alloy in NaCl-KCl-MgCl₂ mixture salt at 650°C. Heat treatment and electrochemical method were used as salt purification methods. An elemental analyzer was used to analyze the O and H contents to measure the amount of water compounds in the salts. Water was quantitatively introduced in the form of H₂O using MgCl₂ hexahydrate.

For salt purification, heat treatment and electrochemical method were applied. Heat treatment was performed on the salts at 200°C for 24 hours, at 200°C and 400°C for 24 hours each, and at 200°C, 400°C, and 600°C for 24 hours each to determine the optimal treatment conditions based on time. For electrochemical method, the H₂ and O₂ evolution potentials are investigated through cyclic voltammograms and then applied to remove O and H from the salts. As-received and optimally heat-treated salts melted at 500°C, and electrochemical method was conducted to compare efficiency with heat treatment.

After purification, immersion corrosion tests were performed in all kinds of salts, and the corrosion speed by water contents was compared in mass change. Morphology and corrosion damages were observed using SEM and EDS. The used salts were analyzed with ICP-OES to confirm the dissolution of material elements and changes in composition.

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Keywords: Molten salt reactor, Salt purification, Corrosion, Heat treatment, Electrochemistry

Preliminary Study on Analytical Methods for Fe and Ni in Radionuclide Analysis of Radioactive Waste

Homyung Lee, Eunmi An, and Yeong su Ha*

Korea Research Institute of Decommissioning (KRID)

*zerowater@krid.re.kr

For the disposal of radioactive waste generated during the operation and decommissioning of nuclear power plants, it is necessary to quantify the concentrations of 14 specified radionuclides, including gross alpha, according to the radioactive waste acceptance criteria. When various radionuclides are present in radioactive waste, they may cause mutual interference, hindering accurate analysis. Therefore, separation of the target radionuclides from the waste matrix prior to analysis is essential. In this study, we conducted a preliminary investigation using samples to develop a KRID radiochemical analysis procedure for Fe-55 and Ni-63, which are among the specified key radionuclides in radioactive waste.

The samples were prepared by adding Fe-55 and Ni-63 isotopes to triple-distilled water, and Fe-55 and Ni-63 were separated using anion-exchange resin, Ni (DMG) resin, and a prefilter in DIBK resin. The recovery of the separated Fe and Ni was verified using an Inductively Coupled Plasma Optical Emission Spectrometer (ICP-OES), and measurements were carried out using a Liquid Scintillation Counter (LSC). As a result, satisfactory recovery rates and measurement values were obtained.

Acknowledgements

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Keywords: Radioactivity analysis, Radioactive waste, Fe-55, Ni-63

Development of a Monitoring and Detection System for Sodium Leaks in Sodium-Cooled Fast Reactors (SFR)

Beom Kyu Kim*, Geun Dong Song, Seung Heon Baek, Jeong Ho Hwang, Yeong Ho Son, and Jae Hun Lee

FNC Technology Co., Ltd.

*withbum77@fnctech.com

Sodium-cooled fast reactors (SFRs), which use sodium (Na) as a coolant, present a significant advantage over traditional pressurized light water and heavy water reactors by mitigating the issue of radioactive waste. SFRs achieve this by utilizing fast neutrons to burn high-level radioactive materials generated through nuclear fission. However, ensuring the safety of SFRs remains a critical challenge, necessitating the development of advanced safety and verification systems. Key to this process is the close monitoring of high-temperature materials and structural integrity, along with the verification of overall safety.

In-service inspection (ISI) of power plants typically involves the regular visual inspection of essential components, devices, and welded areas in pipes to detect physical damage, such as cracks and corrosion, and to assess the condition of test surfaces. This process is vital for identifying vulnerable areas and evaluating structural soundness. For pressurized water reactors, inspections are often carried out using optical visual methods or ultrasonic testing of structural materials like reactor coolant pressure boundaries, pipes, and nozzles.

However, SFRs pose unique challenges due to the opaque nature of sodium, which prevents the use of standard visual inspection techniques. Furthermore, electromagnetic inspection methods are limited by the properties of sodium in its molten state. Currently, ultrasonic testing (UT) is considered the only effective technology for visualizing structural materials in SFRs. However, further research is required to understand the use of ultrasonic waves in high-temperature liquid sodium environments, ensuring both technical safety and operational viability. One of the primary challenges is ensuring the durability and reliable performance of ultrasonic transducers under such extreme conditions.

This study focuses on developing an ultrasonic sensor and a C-Scan system for use at room temperature and underwater, which is then adapted for high-temperature sodium environments. The design and optimization of the ultrasonic sensor and system were specifically aimed at ensuring stable performance in the high-temperature conditions of SFRs. To facilitate testing in such environments, a C-Scan system was constructed inside a glove box for use with high-temperature sodium. In the final phase of this study, C-Scan images were produced and visualized to confirm the system's performance at elevated temperatures.

The results of this research contribute to the advancement of ultrasonic flaw detection technology in sodium-cooled fast reactors, ensuring better safety and monitoring capabilities. This work serves as a foundation for the continued development of durable sensors and imaging systems capable of operating in the extreme conditions required for SFRs.

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Keywords: Sodium-cooled Fast Reactors (SFRs), Safety and monitoring systems, Ultrasonic Testing (UT), High-temperature sodium, C-scan imaging

Activity Standardization of ^{55}Fe by the Triple-to-Double Coincidence Ratio Method

Agung Agusbudiman^{1,2,3,*}, Kyoung Beom Lee^{1,2}, Jong Man Lee^{1,2}, Byoung Chul Kim², Dong Hye Heo²,
and Tae Hee Kim²

¹University of Science and Technology (UST)

²Korea Research Institute of Standards and Science (KRISS)

³National Research and Innovation Agency (BRIN)

*agung@kriss.re.kr

The measurement of iron-55 (^{55}Fe) is important for evaluating radioactive waste, particularly in the first few years after nuclear reactors are decommissioned. This is because ^{55}Fe is one of the primary contributors to radioactive waste, owing to the high iron content in steel and other materials used in reactors. With a half-life of 2.737 years, ^{55}Fe decays through electron capture to stable ^{55}Mn , while emitting Auger electrons and low-energy X-rays. Due to its decay characteristics, measuring this radioactive isotope is quite challenging. Therefore, a reliable standard for measurement is essential.

This study aims to develop a ^{55}Fe standard that can be used as a reference for the measurement of ^{55}Fe in the field. This standard provides the measurement traceability, which helps ensure the accuracy of the measurement.

To develop this standard, we performed an absolute measurement of ^{55}Fe activity using the triple-to-double coincidence ratio (TDCR) method which is based on the liquid scintillation counting technique. Five liquid scintillation samples were prepared by gravimetrically adding the ^{55}Fe active solution into a 20 mL high-performance glass vial. In this study, we used 10 mL of Ultima Gold® as the cocktail solution for each sample. Each sample was measured with the KRISS TDCR system, and the efficiency value was calculated using an in-house code developed based on the decay data of ^{55}Fe . The probability values used in the code were calculated using the Monte Carlo code of PENELOPE ver. 2018.

The experimental results show that the final activity value of ^{55}Fe is (100.3 ± 0.8) Bq/mg as of its reference date. This study also investigated the impact of the ionization quenching parameter (kB) and PMT asymmetry on the final activity value. The results indicate that their influence is less than 0.5%.

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Keywords: LSC, TDCR method, ^{55}Fe , Activity standardization

Comparison of Wet Oxidation and Combustion Methods for the Analysis of Tritium and ^{14}C in Concrete Samples

Sora Choi, Kyungwon Suh, and Jung Bo Yoo*

Korea Atomic Energy Research Institute (KAERI)

*sorachoi@kaeri.re.kr

Tritium and ^{14}C are key radionuclides released during the decommissioning of nuclear power plants, and their accurate analysis is essential for the safe disposal of radioactive waste. Concrete is a common material in nuclear facilities, often retaining these radionuclides within its structure, requiring precise analysis methods to assess their concentrations. This study aims to compare the recovery rates of tritium and ^{14}C in concrete samples using two common methods: wet oxidation and combustion. Wet oxidation decomposes the sample chemically, while combustion involves thermal degradation at high temperatures.

In this experiment, simulated concrete samples were analyzed, and the results showed that wet oxidation achieved a recovery rate of 102% for tritium and 95% for ^{14}C . In contrast, combustion yielded a recovery rate of 88% for tritium and 100% for ^{14}C . These findings suggest that wet oxidation is more effective for extracting tritium, while combustion is better suited for ^{14}C analysis, as the high temperature facilitates the conversion of ^{14}C into carbon dioxide for accurate measurement.

While the results align with previous studies, it is important to note that this experiment used simulated concrete samples. Therefore, the trends observed may not fully apply to actual decommissioned nuclear waste, where tritium and ^{14}C may be more strongly bound within the concrete matrix. Wet oxidation may not fully extract these radionuclides in such cases, especially if they are deeply embedded in the material. Thus, the selection of the appropriate method depends on the specific characteristics of the waste material being analyzed. High-temperature combustion may be necessary for materials where the radionuclides are tightly bound.

This study contributes to the ongoing development of effective methods for the analysis of radionuclides in nuclear waste, emphasizing the need for tailored approaches based on the sample type and the physical characteristics of the radionuclides involved.

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Keywords: Tritium, Carbon-14, Combustion method, Wet oxidation

Optimization of Combustion Temperature for the Analysis of Tritium and ^{14}C

Sora Choi, Mun Ja Kang, and Jung Bo Yoo*

Korea Atomic Energy Research Institute (KAERI)

*sorachoi@kaeri.re.kr

Tritium and ^{14}C are critical radionuclides in nuclear decommissioning and environmental radiological studies, necessitating precise analysis techniques for accurate assessment and management of radioactive waste. The objective of this study was to optimize the recovery rates of tritium and ^{14}C from charcoal filter samples using the combustion method, with a particular focus on the effect of combustion temperature on recovery efficiency. Previous studies have demonstrated that higher combustion temperatures facilitate the conversion of ^{14}C into carbon dioxide, improving recovery rates and ensuring more accurate measurements.

This study compared the recovery rates of tritium and ^{14}C at two combustion temperatures: 600°C and 900°C. At 600°C, the recovery rate for tritium was 88%, while ^{14}C showed a lower recovery rate of 77%. In contrast, when the combustion temperature was increased to 900°C, the tritium recovery rate improved slightly to 89%, while ^{14}C exhibited a significant increase in recovery, reaching 97%. The results indicate that higher combustion temperatures lead to a more complete conversion of ^{14}C to CO_2 , which enhances the overall accuracy of the analysis.

Additionally, post-combustion analysis revealed that the background count for ^{14}C after combustion at 600°C was 218 CPM, whereas at 900°C, it drastically decreased to 17 CPM. This substantial reduction in background counts suggests that higher combustion temperatures not only improve radionuclide recovery but also contribute to the cleaning of the combustion tube, reducing contamination and ensuring more reliable results. A mass reduction of 99% in the samples further confirmed that combustion at 900°C led to nearly complete decomposition of the charcoal filter material.

In conclusion, this study demonstrates that higher combustion temperatures are more effective in enhancing the recovery rates of both tritium and ^{14}C from charcoal filter samples. Moreover, high-temperature combustion reduces residual contamination in the combustion system, improving the precision of radionuclide analysis. These findings provide important insights into optimizing combustion conditions for radionuclide analysis, contributing to better radioactive waste management and environmental monitoring practices.

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Keywords: Tritium, Carbon-14, Combustion method, Pyrolysis

Bayesian Approach to Periodic Validation of Scaling Factors

Wook Sohn*, Jieun Park, Sooyeon Lim, and Kahee Jeong

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*wooksohn@khnp.co.kr

To ensure the safe disposal of radioactive waste, regulators require waste producers to assess the amounts and concentrations of radionuclides—referred to as the radioactive inventory—within the waste. This is crucial for the long-term safety of disposal facilities. However, many radionuclides do not emit easily detectable gamma radiation, making them difficult to measure (DTM) without complex radiochemical separation. As a result, direct measurement from outside waste packages is impractical, particularly for nuclear power plants that generate large quantities of waste.

In response, many countries, including Korea, allow the use of an indirect method known as the scaling factor method. This method estimates the inventory of DTM nuclides by measuring their concentration ratios, or scaling factors, in relation to easy-to-measure (ETM) nuclides like Co-60 and Cs-137, which emit gamma radiation. This approach is based on observed correlations between DTM and ETM nuclide concentrations within specific waste streams at nuclear plants.

Once scaling factors are determined for each DTM-ETM pair through radiochemical analysis of representative samples, their validity must be periodically reviewed. In Korea, this validation is required every two years, covering thirteen nuclides (e.g., H-3, C-14, Ni-63) and gross alpha. To validate, scaling factors from newly collected waste samples are compared to existing scaling factors. If all new scaling factors fall within an acceptable range—typically a factor of ten—the existing factors remain applicable. However, there is no clear guidance on how many representative samples are needed for validation, and outliers complicate the process.

If some calculated scaling factors deviate from the acceptable range, the outliers must be evaluated. They can either be included, if proven to be valid, or excluded by demonstrating that they result from errors or are statistically significant outliers. However, this practice can be misleading, as outliers may occur naturally even in random samples, and excluding them based on their appearance as outliers may lead to incorrect conclusions.

A more robust validation method compares the mean values of the new scaling factors to the established ones, rather than focusing on individual outliers. This is because the existing scaling factors are themselves averages derived from a larger set of data over time.

This study demonstrates the effectiveness of a Bayesian method, which can generate distributions of mean scaling factors from new waste packages. The results indicate that even when outliers are present, this method allows for a more accurate determination of whether the existing scaling factors are still valid. Additionally, the Bayesian approach can help estimate the minimum number of samples needed for future scaling factor calculations.

Keywords: Bayesian inference, Scaling factor, Outlier, Radioactive waste, Inventory assessment

Colorimetric Quantification of Eu(III) in Superheated and Pressurized Aqueous System

Ranyeong Choi and Jun-Yeop Lee*

Pusan National University (PNU)

*jylee@pusan.ac.kr

For the reliable safety assessment of a high-level waste (HLW) repository, it is crucial to obtain chemical thermodynamic data for various radionuclides that accurately reflect the actual deep geological disposal environment. However, in certain HLW disposal scenarios, the temperature in the vicinity of the repository can rise above 100°C due to the decay heat of radionuclides. Although it is highly important to understand the chemistry of radionuclides under superheated ($T > 100^\circ\text{C}$) aqueous conditions, there is a significant lack of available data due to the practical difficulties of conducting experiments under these conditions.

The objective of this study is to perform in-situ quantitative analysis of Eu(III) using a colorimetric approach in superheated ($T = 170^\circ\text{C}$) and pressurized (10 MPa) aqueous condition. The optical cell for colorimetric analysis was made from titanium alloy (Ti-6Al-4V, Grade 5) equipped with sapphire windows providing an optical path length of 2.32 cm. The temperature of system was controlled by custom-made cartridge heaters and K-type thermocouple and the pressure control was achieved with a pressure regulator connected to an Ar gas cylinder. In-situ colorimetric analysis were conducted with UV/Vis absorption spectrometer (Analytik Jena, SPECORD S 600). For the preparation of the chloride-free Eu(III) stock solution, $\text{Eu}(\text{OH})_3(\text{am, fresh})$ was synthesized in Ar-filled glovebox ($\text{O}_2 < 5 \text{ ppm}$) by adding an excess amount of NaOH solution to the EuCl_3 stock solution. After that, the $\text{Eu}(\text{OH})_3(\text{am, fresh})$ solid phase was washed five times with deionized water and subsequently dissolved in HClO_4 solution. Finally, three different Eu(III) concentrations of 8, 10, and 12 mM were prepared by diluting the Eu(III) solution with 0.1 M HClO_4 solution.

The results obtained in the superheated and pressurized aqueous system clearly revealed the characteristic absorption peak of Eu(III) at 394 nm with a molar absorptivity of $\epsilon = 1.34 \text{ M}^{-1} \cdot \text{cm}^{-1}$. The absorption peak position of Eu(III) under superheated water condition corresponds well with that measured at room temperature. However, the molar absorptivity in the superheated water environment showed a notably lower trend compared to the value obtained at room temperature ($\epsilon = 2.26 \text{ M}^{-1} \cdot \text{cm}^{-1}$). The temperature-dependent molar absorptivity might be explained by changes in the energy levels and transition probabilities of the orbital electrons of Eu(III). The optical cell and spectroscopic system established in this study are expected to be further employed to investigate the aqueous chemistry of lanthanides and actinides using various spectroscopic tools under superheated/pressurized aqueous conditions.

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Keywords: Superheated water, Eu(III), UV/Vis, Colorimetry

Development and Verification of a Radiometric Analysis Method for Sn-113 in RI Waste From Non-Nuclear Industries

Sangjun Han¹, Bogil Kim^{1,*}, Yongju Shin¹, Hyegyeong Ha¹, Junseok Kim¹, Wooju Hong¹, Taeyong Noh²,

and Daehee Kim²

¹RMTEC Co., Ltd.

²WITHTECH

*bg1123@rmtec.co.kr

This study focuses on developing and verifying a precise radiometric analysis method for tin-113 (Sn-113) found in radioactive isotope (RI) waste generated from non-nuclear industries and medical fields. Sn-113 is a radionuclide with a half-life of about 115 days that emits gamma radiation, posing potential radiation exposure risks to humans and the environment. However, due to the diversity and low occurrence of RI waste, existing nuclide analysis techniques are insufficient to meet the demand for the quantitative analysis of Sn-113. Therefore, this study aims to enhance the safety of RI waste management by developing a new analytical method for Sn-113 and validating the procedure using standard materials.

During the study, Sn-113 was separated using anion exchange resin and TRU resin. Interfering nuclides were removed using HCl, and tin was eluted using HNO₃ and HCl. The eluate underwent further purification using TRU resin with HCl, followed by final elution of Sn using HNO₃. The analysis was performed using ICP-MS, ICP-OES, and HPGe detectors, and Sn-118 was used as a tracer to measure recovery rates. The proposed analytical method can contribute significantly to the accurate detection of Sn-113, a difficult-to-analyze nuclide in RI waste, and to monitoring its presence in radioactive waste.

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Keywords: Sn-113 analysis, Radioactive isotope waste, Ion exchange resin separation, ICP-MS and HPGe analysis, Medical and non-nuclear industry waste management

한국원자력환경공단(KORAD)
고준위방폐물관리 전문인력 양성
특별세션(학생 Poster)

Criticality and Shielding Assessment for Deep Disposal Facilities Based on Horizontal Disposal

Nak Woong Yang*, Min Sik Jeon, Ji Hwan Ma, Seok Geun Cho, and Chang-Je Park

Sejong University

*skrdnd1028@gmail.com

Due to the high levels of radioactivity and heat generation, spent nuclear fuel requires temporary storage in wet storage pools located within nuclear power plant sites. However, with the saturation of these storage pools anticipated at several sites, including the predicted Kori Unit 1 decommissioned, the establishment of a high-level radioactive waste disposal facility becomes essential. This is expected to be implemented as a deep disposal facility.

The purpose of this study is to design a canister suitable for domestic nuclear fuel types by referencing the STAD canister used in the United States and to evaluate the radiological safety from assessing the criticality and shielding within the disposal facility. The advantage of horizontal disposal is that it eliminates the need for the rotation process required for vertical disposal and minimizing the ground disturbance that would occur with vertical deposition holes. This paper simulates that modeling the canister and disposal facility using the MCNP6 and ORIGEN-S codes.

In the critical evaluation, when the storage canister loaded with new fuel was flooded, the $k_{\text{eff}} = 0.87717$ met the design target even under conservative conditions. The shielding evaluation employed the WH 14×14 type of nuclear fuel, with a cooling period of 10 years. The radiation dose of 0.03174 mSv/h at the top of the cask and 1.00084 mSv/h at the side was determined to be below the threshold of 2 mSv/h on the surface of the spent fuel storage cask. Furthermore, the shielding was assessed at three locations within the tunnel where the four containers were situated. The highest recorded value was 1.6177E-02 mSv/h, which met the design objective of 0.05 mSv/h for the dose rate within the tunnel.

However, it is necessary to evaluate the structural and thermal stability of the disposal canister and optimize the method of disposition canister in the disposal facility because the deformation over time of the disposal canister is not considered. Unlike conventional vertical disposition, this study explains the benefits of horizontal disposal method and presents the possibility of horizontal disposition in future in deep disposal facilities.

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Keywords: Spent nuclear fuel, Deep disposal, Horizontal disposal, Criticality, Shielding

Scenario and Modelling of Concrete Deterioration Using GoldSim for Radioactive Waste Management

Yong Hee Han*, Sang Ryeol Yoon, Dong Geon Jang, Jun Yeong Park, Hyun Jin Yu, Ki Tae Yang,
and Jong Soon Song
Chosun University
*rfvg102@chosun.kr

Concrete is crucial in domestic high-level radioactive waste disposal environments, serving as a key engineering barrier within the multi-barrier concept for radioactive waste containment. Over time, concrete can deteriorate due to factors such as waste dissolution, groundwater flow, subsidence, and seismic activity, potentially leading to the release of waste into the environment. A common deterioration mechanism is sulfate attack, especially when concrete structures interact with groundwater.

Ensuring a high-level radioactive waste disposal facility's safety post-closure requires analyzing potential phenomena within the facility and its surroundings. While direct prediction of these phenomena is challenging, simulation programs can evaluate predictable scenarios. Among various safety assessment programs, GoldSim stands out. It was instrumental in licensing the Yucca Mountain repository in the US and excels in simulating mass and volume transfer in complex disposal systems, including nuclide migration and groundwater flow. GoldSim has also been extensively used to study concrete deterioration.

Before creating scenarios in GoldSim, a preliminary study examined concrete deterioration due to sulfate infiltration using KURT groundwater. An experimental tank was constructed, and 20-year-old waste concrete was prepared, cut into 36 specimens of $\varnothing 50 \times 50$ H sizes, and classified into grades A to D. The specimens were immersed in a water tank for 365 days to accelerate deterioration. Every 30 days, specimens were removed for analysis using SEM and EDS to observe deterioration phenomena. The experimental data were then applied to GoldSim to model the impact on the ecosystem runoff process.

This study outlines a basic evaluation model using GoldSim, incorporating data from preliminary experiments. A scenario was modeled where radioactive waste leaks into the ecosystem through a series of barriers: disposal container, silo, concrete barrier, and natural barrier. The results highlight the influence of concrete deterioration on the release of radioactive waste into the natural ecosystem.

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Keywords: Goldsim modeling, Concrete deterioration, HLW, Radioactive waste, Post-closure safety assessment, FEP

Feasibility of Time-Series InSAR for Monitoring the Stability of Underground Disposal Facility

Joonbeoum Park*, Seonjae Yoo, Jin Chon, Taewook Kim, Hyangsun Han, and Woodon Jeong

Kangwon National University

*pjb10241@naver.com

Underground disposal facilities are designated for the geological disposal of radioactive waste or the storage of carbon dioxide. The operation of these systems can cause surface displacement, influenced by various factors such as geological characteristics, the amount of disposed material, and meteorological conditions, which significantly impact the stability of the facilities. Therefore, monitoring surface displacement above underground disposal sites is essential for assessing their stability and ensuring effective management. Time-series satellite synthetic aperture radar (SAR) interferometry has been widely used to measure surface displacement in various research fields. In this study, we applied a time-series interferometric SAR (InSAR) technique to 172 Sentinel-1 SAR images, acquired from 2018 to 2023, to measure surface displacement around an underground disposal facility in Shanxi, China. The displacement time series was retrieved from both permanent scatterers and distributed scatterers, with a total of 3,865 scatterers identified around the facility. Since the facility's operation began in June 2020, a maximum uplift of approximately 60 mm, at a rate of ~ 10 mm/year, was observed. This uplift was attributed to the injection of carbon dioxide into the underground system. This study demonstrates that time-series InSAR monitoring is a powerful tool for analyzing surface displacement and evaluating the stability of underground disposal facilities.

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Keywords: InSAR, Surface displacement, Stability, Underground disposal facility

Improving Seismic Imaging Using Unsupervised Anomaly Detection

Chanha Park*, Jaehoon Cho, Junhyun Kim, Chanhee Lee, and Woodon Jeong

Kangwon National University

*201913898@kangwon.ac.kr

Seismic exploration and imaging are crucial to deep geological disposal of high-level radioactive waste, where accurate imaging of subsurface structures is critical for safe and effective site investigation and monitoring. Especially, common midpoint (CMP) stacking is a widely used and powerful imaging method for delineating subsurface structures, playing a crucial role in site investigation by generating preliminary images of the subsurface. Traditional CMP stacking assumes that seismic events are accurately aligned after normal moveout (NMO) correction, given precise velocity information. This alignment results in well-focused wavelets at each subsurface reflector. However, in practice, velocity information is often imperfect and noisy, leading to misaligned seismic events that degrade the quality of the stacked image. This study introduces a novel approach to enhancing CMP stacking by removing inaccurately corrected events and extra noise components through unsupervised anomaly detection. Specifically, the local outlier probabilities (LoOP) algorithm, which is a branch of unsupervised machine learning, assesses the likelihood of each data point being an outlier within a dataset. By applying LoOP to seismic data before stacking (e.g., to NMO-corrected CMP gathers), the method produces clearer and sharper seismic images by mitigating the adverse effects of misaligned events, which often act as outlying objects. A key advantage of the proposed method is its ability to minimize human intervention in determining parameters for distinguishing signal from noise. In this study, the LoOP algorithm was applied to a simple 1D model, a synthetic dataset with incorrect velocity information, and field data from a reservoir exploration in Norway. Assuming that sample data contains more source data than noise, the results demonstrated significant improvement in data quality, confirming the effectiveness of LoOP in both synthetic and field data applications. This suggests that LoOP can greatly enhance seismic data interpretation by effectively separating noise components from the desired signal components.

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Keywords: Local Outlier Probability (LoOP), Unsupervised machine learning, Seismic data, Noise reduction

Fractures Development History and Paleostress Change in the Gijang Coastal Outcrops

TaeYong Hyun*, Minkyu Kim, and Young-Seog Kim

Pukyong National University

*hty511@pukyong.ac.kr

As interim storage facilities for high-level radioactive waste, including spent nuclear fuel from nuclear power plants, are expected to reach capacity by 2030, the need for permanent disposal sites for high-level radioactive waste is increasing. Since these disposal sites must safely isolate high-level radioactive waste for the next 100,000 years, the chosen site should be geologically stable, with low potential for structural deformation and low permeability. Fractures developed in bedrock may be reactivated due to changes in the stress field, and in low-permeability bedrock, fluid flow tends to concentrate along faults and fractures. For this reason, analyzing the fracture development history and paleostress field within the bedrock of the disposal site is of utmost priority. In this study, we analyzed the fracture development history and paleostress field in a granitic area near the Ilgwang Fault. The study was conducted in the following order. First, we analyzed lineament within a 500-meter radius of the study area. Then, we classified fracture sets to investigate the characteristics of each fracture set and the development history of the area, followed by geometric and cross-cutting relationship analysis for each fracture set. The results show that the lineament striking NW-SE and ENE-WSW dominate the study area. The orientations of the fractures identified in the field investigation are similar to those from the lineament analysis, and based on this, the following fracture sets were classified: (Set A) left-lateral strike-slip faults striking NW-SE, (Set B) right-lateral strike-slip faults striking ENE-WSW, (Set C) fractures striking WNW-ESE, and (Set D) tensile fractures and veins striking E-W. The fracture development history inferred from the cross-cutting relationships is as follows: (1) conjugate fracture sets B, C, and secondary fracture set D, (2) right-lateral strike-slip fault A, which cross cuts the previous fractures. Based on the geometry and cross-cutting relationships of the fractures, the paleostress field was found to have transitioned from E-W compressional stress to WNW-ESE compressional stress. This analysis of fracture development history and the paleostress field can provide important information about geological structures and ground stability, enabling early identification of the long-term stability of the site and the risk of geological hazards. It is therefore essential for evaluating the safety of nuclear power plants and key national facilities. We expect that this study will contribute to site investigations for the construction of nuclear power plants and high-level radioactive waste disposal facilities.

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Keywords: Fault, Fracture, Development history, Paleostress

Calcium Carbonate Minerals for Sr Ionic and Mineral Substitution: Experiments and Quantum Mechanical Modeling

DongHoon Kim*, TaeWoong Son, and YoungJae Kim

Pukyong National University

*terrific0806@naver.com

After nuclear power generation, radioactive elements such as strontium-90 (^{90}Sr) are predominantly produced in spent nuclear fuel and nuclear waste. Strontium-90 is a pure beta emitter with a half-life of 28.8 years, undergoing beta decay to produce Yttrium-90 (Y-90) (Park et al., 2009). Due to its chemical similarity to calcium (Ca), ^{90}Sr tends to accumulate in bones and teeth, posing significant health risks such as bone cancer and leukemia. Given its long half-life, ^{90}Sr can cause prolonged harm to human health.

If released during the disposal of radioactive waste, ^{90}Sr is likely to remain dissolved in fluids, making it a radionuclide that requires careful monitoring in groundwater and surface water. However, once radioactive materials like ^{90}Sr enter the natural environment, they are challenging to remove, and current solutions are limited. Therefore, developing cost-effective and efficient techniques for strontium removal using natural materials is crucial.

One promising approach involves the use of naturally occurring minerals for the immobilization of dissolved strontium. This method is reported to be simple, cost-effective, and environmentally friendly, providing a long-term solution for managing radioactive contaminants (Kohei Tokunaga et al., 2018).

This study was divided into two main areas. First, density functional theory (DFT) calculations were used to evaluate the stability of strontium substitution by examining the exchange of calcium with strontium in natural carbonate minerals—specifically, the polymorphs of calcium carbonate: calcite, aragonite, and vaterite. Second, experiments were conducted to compare the strontium removal efficiency across different polymorphs by adjusting variables in the mineral replacement reactions. Our research demonstrates calcium carbonate minerals as promising materials for the immobilization of dissolved strontium ions from spent nuclear wastes.

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Keywords: Strontium, Immobilization of radioactive elements using minerals, Density Functional Theory (DFT) calculation, Mineral replacement

Establishing a Social Safety Net for High-Level Radioactive Waste

Seo Jin Choi*, Min Ji Kim, Jun Hee Kim, Yeon Jung Ryu, Eun Gyeong Park, Hea In Shin, Yeo Jin Jeong,

Min Woo Jeon, Jung Hoon Kim, and Yeon Hee Kang

Catholic University of Pusan

*ttngss67@gmail.com

In our country, the amount of high-level radioactive waste (HLW) in the form of spent nuclear fuel is expected to increase by approximately 150,000 bundles compared to previous estimates, which will likely advance the saturation point by about 1 to 2 years. However, local residents are resisting the construction of temporary storage facilities for managing this HLW, and there is currently no legal foundation in place to support it. Social safety nets related to HLW can be considered as social response systems that ensure the safety of both the community and individuals.

A literature review related to the management of HLW was conducted to develop a survey tool. After a preliminary survey was carried out to check the respondents' understanding of the questions and their ability to respond, an online survey was conducted over 18 days targeting residents in the Busan. The results from 204 respondents were analyzed using IBM SPSS Statistics version 28.0. To test for differences between variables, t-tests and one-way ANOVA were performed, and Pearson correlation analysis was conducted to examine the relationships between variables.

This study confirmed that local residents have a high perception of the risks associated with high-level radioactive waste (HLW), but they also expressed hope that these risks would be reduced through government efforts. Additionally, it was found that as the expertise of HLW managers increases, HLW management and safety training are conducted more effectively. Therefore, for the establishment of a social safety net for HLW, it is crucial to ensure both government efforts and the expertise of managers responsible for its management.

Therefore, if organizations or government bodies put efforts into effectively establishing a social safety net for high-level radioactive waste (HLW), it is believed that this will reduce citizens' anxiety about HLW and positively influence government policy decisions.

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Keywords: High-Level radioactive waste (HLW), Social safety net, Risk perception, Safety training

Establishing a Safety Culture for High-Level Radioactive Waste

Hyo Won Lee*, Jun Young Kwon, Ri Yun Kim, Su A Park, Si Eun Park, Seong Min Shin, Ji Hyeong Lee,

Chae Yeon Lee, Sung Hee Yang, and Yong In Cho

Catholic University of Pusan

*kyhlhw2@naver.com

The study was conducted to investigate the impact of citizens safety awareness in regions with nuclear power plants on the establishment of high-level radioactive waste (HLW) safety culture. A survey was administered to 176 citizens in Busan, a region where there is a possibility of constructing an HLW dry storage facility. The survey consisted of questions on safety awareness and HLW safety culture. Correlation analysis and multiple linear regression analysis were used to examine the relationships and influences of central variables.

The results are as follows:

1. Lower age, education level, and monthly income were associated with higher citizen safety awareness and a better perception of the establishment of high-level radioactive waste (HLW) safety culture.
2. Among the sub-factors of citizen safety awareness, safety education and understanding were found to be the most influential in the establishment of HLW safety culture.
3. It was observed that better implementation of safety regulations for HLW led to smoother communication with citizens.

If future research is conducted across more regions on the impact of citizen safety awareness on the establishment of high-level radioactive waste (HLW) safety culture, it is believed that a variety of findings on HLW safety culture establishment could be obtained.

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Keywords: High-level radioactive waste, Safety culture, Citizen survey, Citizen safety consciousness, Regional research

The Impact of Residents' Self-Determination on Coping Behavior in Radioactive Waste Disposal Facility Siting: An Analysis of the Moderating Effect of Social Capital

Ji-Hwan Oh*, Seong-Hyeon Kim, Ha-Eun Lee, Dong-Jin Jeong, and Eun-Young Hong

Ajou University

*wlgks12345@naver.com

This study aims to analyze the social conflicts that arise during the siting process of radioactive waste disposal facilities as nuclear power generation is gaining attention as a response to climate change. According to the IPCC Sixth Assessment Report (2021), greenhouse gases emitted by human activities have increased global surface temperatures by 1.1°C compared to 1850-1900. Furthermore, it is predicted to rise to 1.5°C in the near future (2021-2040) if greenhouse gas emissions continue at the current rate. As a result, climate change is causing extreme weather events such as flooding, heavy rainfall, desertification, and typhoons, causing severe damage. The international community has adopted several agreements to combat climate change, but there are still limitations in reducing the use of fossil fuels. Under these circumstances, nuclear power has been reemerging as a viable alternative to achieve carbon neutrality, and many countries are shifting from anti-nuclear to pro-nuclear policies.

However, the expansion of nuclear power generation is accompanied by the issue of high-level radioactive waste disposal, which has led to serious conflicts between local communities and governments during the siting process. In South Korea, in particular, the current temporary storage facilities at nuclear power plants are expected to be saturated by 2030, and there is an urgent need to discuss the construction of a repository to permanently dispose of high-level radioactive waste. However, the construction of a repository is difficult to agree upon among stakeholders due to its inherent risks and social backlash, and conflicts persist.

This study aims to analyze the impact of residents' self-determination on their response behaviors (individual, community, and policy) in this conflict situation and to identify how social capital moderates this relationship. The research model is based on the Theory of Planned Behavior (TPB) and the risk perception paradigm, and includes attitudes, personal norms, social norms, perceived control efficacy, perceived risk, perceived benefit, subjective knowledge, objective knowledge, and negative emotions as independent variables. Self-determination is set as the variable that mediates the effects of these independent variables on residents' coping behavior, and social capital is the variable that moderates the relationship between self-determination and coping behavior.

This study is expected to contribute to providing policy implications for resolving social conflicts that arise during the siting process of radioactive waste disposal facilities and enhancing residents' participation and trust. This study aims to provide important data for understanding social conflicts related to the expansion of nuclear power generation in response to climate change, and for developing strategies to resolve them.

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Keywords: Radioactive waste disposal, Nuclear power generation, Social conflict, Self-determination, Theory of Planned Behavior (TPB), Social capital

A Study on the Effectiveness of Nuclear Energy-Related Education: Application of the Pre-Post Control Group Comparative Experimental Method

Yun-Ah Pi*, Sih-Yein Lee, Sun-young Lee, Hyo-I Yoo, and Hye-Ji Han

Ajou University

*puna0818@naver.com

In recent years, nuclear power has gained increasing attention as a means of ensuring a stable energy supply. Nuclear power has a high degree of energy technology independence compared to other energy sources, and is considered an important alternative for national energy security and sustainable development. Accordingly, domestic governments and nuclear-related organizations are making various efforts to promote the social acceptance of nuclear power. However, the revitalization of nuclear power generation requires not only the technical aspects, but also the understanding and cooperation of citizens on social acceptance and waste disposal issues.

Existing studies discussing means to increase nuclear acceptance suggest that education can play an important role, but empirical examination of the impact of the specific content or form of education on cognitive dimensions is lacking. To date, the importance of nuclear education has been emphasized by various organizations and governments, but there is a lack of empirical research on its effectiveness.

Therefore, this study aims to clarify the practical effects of nuclear education by analyzing the impact of nuclear education on cognitive dimensions. A before-and-after control group comparative experimental design was used to analyze the differences in objective knowledge, subjective knowledge, and beliefs about nuclear energy and related issues depending on whether nuclear energy education was provided or not.

The ultimate goal of this study is to empirically analyze the impact of nuclear energy education on cognitive variables such as objective knowledge (what we know), subjective knowledge (what we think we know), and beliefs in false information, and to clarify the practical effects of education. As there may be a gap between the level of acquisition of nuclear knowledge by individuals and the level of knowledge acquisition perceived by individuals, we categorized knowledge in the cognitive dimension into objective knowledge and subjective knowledge.

Therefore, we analyzed how objective and subjective knowledge levels are affected by the effect of nuclear energy-related education by separating the two types. By analyzing group differences in beliefs about the issue, we aimed to determine whether nuclear energy education actually has an impact on improving citizens' ability to critically think about nuclear energy-related issues. This study sought to determine whether nuclear education is an effective means of improving citizens' ability to critically examine and accept nuclear misrepresentations in the media.

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Keywords: Nuclear education, Objective knowledge, Subjective knowledge, Belief in issues (false information)

A Multidimensional Acceptance of High-Level Radioactive Waste Disposal Facility Siting: Focusing on Reluctant Acceptance

Ye-Seo Yang*, Su-Been Park, Hyun-Bi Ahn, Tae-Yeong Kim, and Hye-Ji Han

Ajou University

*ekdud990@naver.com

The continuous economic growth and increasing energy demand have brought fossil fuels to their limits. Furthermore, the global awareness of the climate crisis and the accelerating pace of global warming has heightened the urgency of transitioning to renewable energy sources. However, the variability and technical limitations of renewable energy make it difficult to meet current energy demands. In this context, nuclear energy is perceived as a relatively stable power source, which has brought the issue of high-level radioactive waste management to the forefront. Given that nuclear power has become an essential energy source, this study aims to explore the acceptability of site selection for high-level radioactive waste disposal facilities in depth. The risks associated with a “high-level radioactive waste disposal facility” make it unlikely that many people would willingly accept such a facility. Moreover, existing studies have shown that it is very difficult to bridge the gap between stakeholders’ conflicts regarding acceptance. The concept of “reluctant acceptance”, as presented in this study, refers to a passive acceptance by residents who, despite recognizing the risks of nuclear power and radioactive waste management, have no alternative or are compelled by practical necessity to accept the situation due to external circumstances. Given the specific characteristics of a “high-level radioactive waste disposal facility”, focusing on reluctant acceptance rather than general acceptance provides a more realistic approach. By addressing the limitations of previous acceptance studies that failed to adequately reflect the real-world situation, this research aims to accurately measure the attitudes that people actually perceive.

The objectives of this study are as follows: First, to discuss whether the concept of reluctant acceptance can be appropriately applied to the acceptance of “high-level radioactive waste disposal facility site selection”. Second, to analyze the factors influencing “reluctant acceptance” and understand their structural relationships. Third, to explore whether the independent variables discussed in previous acceptance studies can also be applied to the acceptance of radioactive waste disposal facilities. Through this, the study aims to explore and compare the determinants of acceptance for each “high-level radioactive waste disposal facility” and identify the factors that could help alleviate public resistance or anxiety during the site selection process in South Korea. By providing practically applicable perspectives, this research is expected to contribute to a more realistic understanding of acceptance.

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Keywords: Reluctant acceptance, High-level radioactive waste disposal, Site selection, Public resistance, Risk perception

Research on the Formation Mechanisms of NIMBY (Not In My Backyard) Attitudes: Focusing on the Case of a High-Level Waste Repository

Chae-Eun Lee*, Jeong-Woo Park, Min-Hee Kim, Jun-Hyuk Choi, and Eun-Young Hong

Ajou University

*echoyee_@naver.com

South Korea faces challenges with energy security due to its reliance on imported fossil fuels, which are becoming unstable because of changes in the global political and social environment. At the same time, the international community's aggressive transition to a decarbonized economy presents the country with the need to strengthen its energy security while addressing climate change. Nuclear power is a crucial energy source for South Korea, accounting for about 30% of its electricity generation. However, the management of high-level radioactive waste remains a significant issue. Currently, this waste is stored in temporary facilities that are expected to reach capacity by 2030, creating an urgent need for a high-level waste repository. The government has encountered many difficulties in the process of selecting low- and medium-level repositories due to strong opposition from local residents. Since high-level radioactive waste is highly dangerous and requires deep geological disposal, the phenomenon of Not In My Backyard (NIMBY) is expected to be inevitable. Therefore, convincing local residents is likely to be the biggest challenge in the site selection process for a high-level waste repository. Focusing on NIMBY attitudes, this study aims to investigate the impact of trust in government nuclear energy policies on residents' NIMBY attitudes during the site selection process of a high-level waste repository, analyze the factors that influence the formation of NIMBY attitudes, and draw implications for conflict resolution.

For this purpose, we set individual behavior and risk perception factors as independent variables based on existing research. To measure the dependent variable, NIMBY attitudes, we created a NIMBY level variable by subtracting the response "I favor ('accept') the installation in my area" from the response "Acceptance of high-level radioactive waste storage". Additionally, we constructed a research model incorporating trust in the government's nuclear policy as a parameter and the level of opinion expression as a moderating variable.

The results of a path model showed that technology optimism, experience of energy shortage, perceived benefit, and subjective knowledge had a significant positive effect on trust in government nuclear policy, while growth limits, energy security crisis, and perceived risk had a significant negative effect. The results of the b path model confirmed that trust in government nuclear policy has a negative effect on the NIMBY attitudes. Moreover, the interaction terms between the parameter and the moderating variables showed a significant negative effect, especially the moderated mediation effects of the energy shortage experience, perceived benefit, perceived risk, subjective knowledge, and negative image variables.

This study has theoretical value in that the dependent variable was set in a new way to measure NIMBY attitudes, which distinguishes it from previous research. The verification of the moderated mediation effect through a research model that includes both parameter and moderating variable provides a more in-depth analysis. In addition, by analyzing the factors that shape residents' NIMBY attitudes, this study offers policy significance by providing a basis for establishing communication and cooperative relationships with residents in the siting process of hazardous facilities in the future.

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This paper is a result of The Human Resources Development Project for HLW Management hosted by KORAD.

Keywords: NIMBY (Not In My Backyard) attitudes, High-level waste repository, Trust in government nuclear policy, Risk perception, Site selection process, Energy security crisis

An Exploration of the Factors Influencing Changes in the Acceptance of Nuclear Power Plants and High-Level Radioactive Waste Disposal Facilities

Myeong-Seong Kim*, Jin-Seong Kim, Min-Seok Jang, Seok-Hwa Jeong, and Chang-Geun Yun

Ajou University

*rlaudtjd55@naver.com

This study aims to explore the factors influencing changes in the acceptance of nuclear power plants and high-level radioactive waste disposal facilities (disposal sites), focusing on personal characteristics and perceptions of nuclear policies. The current era is facing various environmental issues due to the increased use of fossil fuels. In efforts to reduce fossil fuel usage, alternative energy production methods, such as using the nuclear fission process in nuclear power plants, are being employed. Nuclear energy plays a crucial role in terms of energy security and environmental protection. However, due to the risks associated with radioactive materials and the possibility of accidents, such as the Fukushima nuclear power plant disaster in Japan, it has become a topic of social debate. In particular, the safe handling of radioactive waste is a key issue in these discussions. Despite this, public opposition, especially from local residents, to the siting of disposal sites has led to policy deadlocks.

First, this study analyzed how personal characteristics such as age, gender, education level, and political orientation influence acceptance of nuclear facilities. Second, it assessed the impact of trust in government nuclear policies, transparency in the policy-making process, and access to information on acceptance. Third, it explored the complex interactions between personal characteristics and perceptions of policy on acceptance. To achieve this, a comprehensive interpretation of data was conducted through national surveys and both statistical and qualitative analyses.

The results of the study indicate that age and political orientation have significant impacts on the acceptance of nuclear facilities. In particular, individuals with strong political orientations tend to have clear positive or negative stances on nuclear facilities. Additionally, higher trust in government nuclear policies is associated with increased acceptance of both nuclear power plants and high-level radioactive waste disposal sites. Transparency and accessibility of information also play an important role in acceptance, and it was confirmed that ensuring these factors can help mitigate opposition.

The study emphasizes that, in order to improve the acceptance of nuclear facilities, it is essential not only to provide information but also to enhance transparency and public participation in the policy-making process. Furthermore, it suggests that tailored policy designs that take into account the characteristics of local communities and individuals are necessary. These findings are expected to provide practical policy implications in the process of formulating nuclear policies and selecting sites for disposal facilities.

Support for or opposition to nuclear power plants and high-level radioactive waste disposal sites was categorized into four groups using an ARNOVA test. The four types are: 1) support for nuclear power plants and disposal sites, 2) support for nuclear power plants but opposition to disposal sites, 3) opposition to nuclear power plants but support for disposal sites, and 4) opposition to both nuclear power plants and disposal sites. Regression analysis was then conducted on each group to analyze causal relationships, and significant results were confirmed by examining whether meaningful differences in the averages of each model could be derived.

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Keywords: Nuclear power plants, High-level radioactive waste disposal facilities, Acceptance, Policy trust, Policy transparency

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발행인 정재학

발행처 한국방사성폐기물학회

주 소 대전광역시 유성구 대덕대로 989번길 111
한국원자력연구원 원자력사업기술지원동(E7) 202호

연락처 Tel +82-42-861-5851
Fax +82-42-861-5852
E-mail krs@krs.or.kr



사단 한국방사성폐기물학회
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