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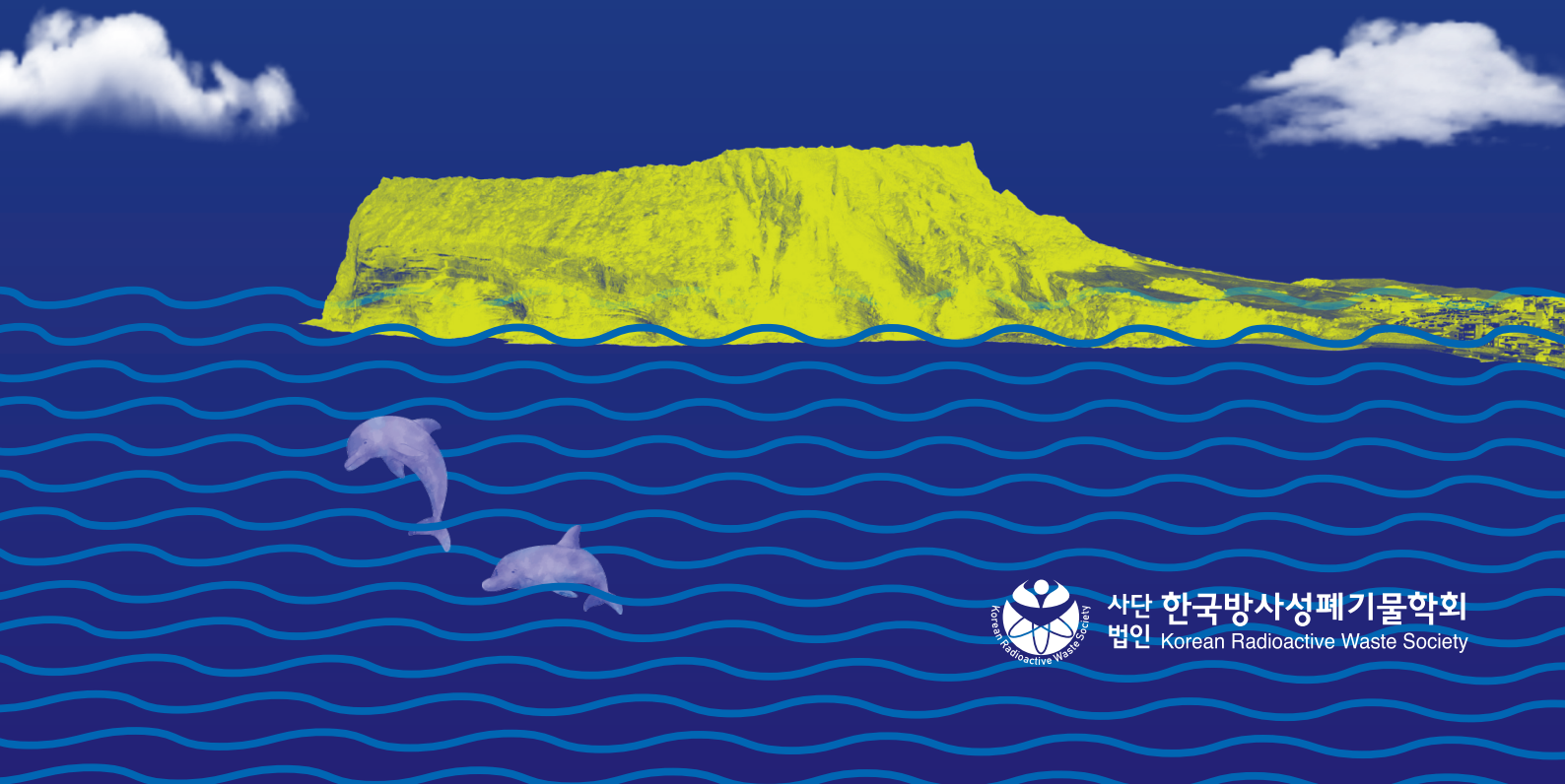
한국방사성폐기물학회
2023 추계학술발표회
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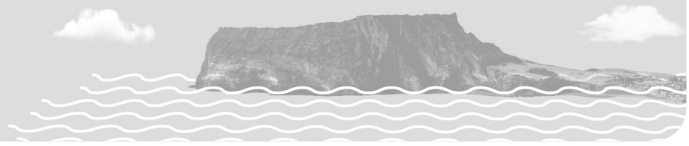
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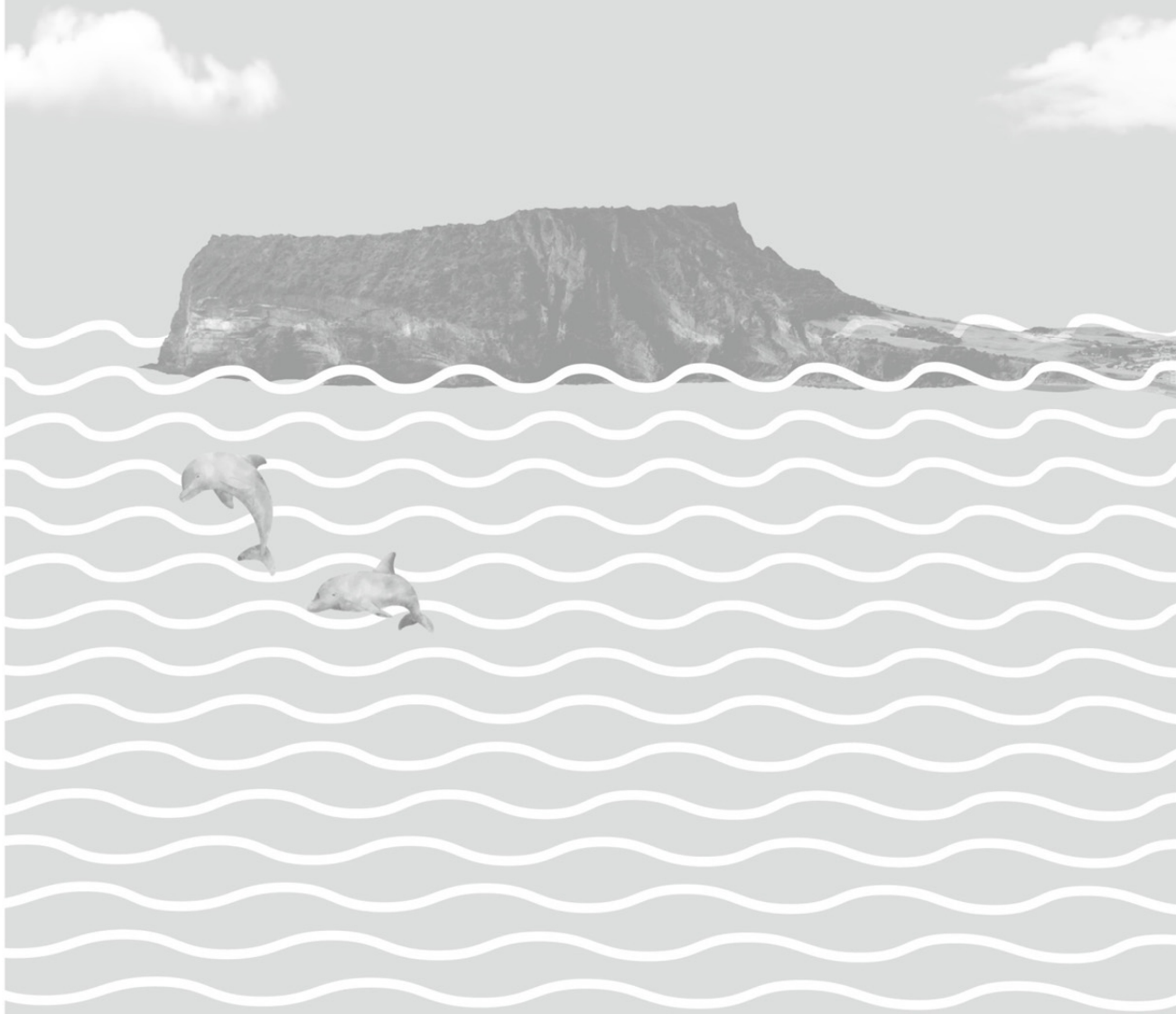
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1분과

핵주기정책·규제 및 비확산 (ORAL)



Consideration on the Relationship Between Security and Safeguards

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SMR, which has recently been in the spotlight, has several advantages. However, it poses additional challenges in the areas of new design, digitalization, security, safety and safeguards. Among them, security refers to measures to protect nuclear materials and facilities from unauthorized access, theft, or destruction. Safeguards refer to measures to prevent the spread of nuclear weapons. The relationship between security and safeguards is complex and constantly evolving. In general, security measures are designed to protect nuclear materials and facilities from physical attack, while safeguards are designed to track and monitor the movement of nuclear materials and prevent them from being used to create nuclear weapons. In some areas security and safeguards work in complementary ways, and in other areas they conflict. But ultimately, finding a balance is what is effective and efficient.

In conclusion, although the security and safeguards of SMRs have different key objectives, they are closely related and must be implemented comprehensively and consistently to ensure the safety of nuclear facilities, the public, and the environment.

In this paper, we investigate how the safety and safeguards of SMR are currently being researched and analyze what difficulties there are when assuming that they are operated as a single interface.

Keywords: Safeguards, Security, 3S

Safeguards Distinction in Water-Cooled SMRs

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Small modular reactors (SMRs) are getting attention as an alternative to fossil fuel power stations due to versatile application and carbon dioxide reduction. Although various types of advanced reactors are being developed, water-cooled SMR will be first deployed on a commercial scale. The International Atomic Energy Agency (IAEA) and regulatory bodies are trying to identify safeguards issues of water-cooled SMRs as the first priority. IAEA begins to develop a safeguards plan by asking for the facility's specification in a given format, a design information questionnaire (DIQ). Then, IAEA periodically performs safeguards activities such as design information verification (DIV) and physical inventory verification (PIV). In this sense, we utilize research and power reactor DIQ for water-cooled SMRs (NuScale, SMART, i-SMR and KLT-40S). Most of the questions are answered with open information. For undisclosed answers, pressurized water reactor (PWR) features are described.

Safeguards issues in water-cooled SMR originate from core modularization. As the nuclear material flows are diversified, the number of safeguards measure will be increased while staff are reduced in SMRs. Instrumentation for safeguards should be developed to reduce worker's fatigue level. Intensive arrangement of fuel assemblies may also need unique devices to secure their visibility or detectability. A transparent floor with a surveillance system or advanced Cherenkov viewing device may be adopted to enhance containment and surveillance.

Meanwhile, some questions could be more elaborate regarding safeguards. First, question #38 cannot confirm the time of occurrence of weapon-grade plutonium for reactor operation. Second, the answers in questions #46 and #49 are primitive to identify a place to generate an undeclared fissile material. Therefore, the current DIQ should be revised to get a detailed burnup report and spatial distribution of neutron flux.

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Keywords: Design information questionnaire, Water-cooled, Small modular reactor, Safeguards by design

Development of Method for Deriving Weak Diversion Scenario in Nuclear Material Accountancy (NMA) System Based on Reinforcement Learning

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Nuclear Material Accountancy (NMA) system quantitatively evaluates whether nuclear material is diverted or not. Material balance is evaluated based on nuclear material measurements based on this system and these processes are based on statistical techniques. Therefore, it is possible to evaluate the performance based on modeling and simulation technique from the development stage. In the performance evaluation, several diversion scenarios are established, nuclear material diversion is attempted in a virtual simulation environment according to these scenarios, and the detection probability is evaluated. Therefore, one of the important things is to derive vulnerable diversion scenario in advance. However, in actual facilities, it is not easy to manually derive weak scenario because there are numerous factors that affect detection performance.

In this study, reinforcement learning has been applied to automatically derive vulnerable diversion scenarios from virtual NMA system. Reinforcement learning trains agents to take optimal actions in a virtual environment, and based on this, it is possible to develop an agent that attempt to divert nuclear materials according to optimal weak scenario in the NMA system. A somewhat simple NMA system model has been considered to confirm the applicability of reinforcement learning in this study. The simple model performs 10 consecutive material balance evaluations per year and has the characteristic of increasing MUF uncertainty according to balance period. The expected vulnerable diversion scenario is a case where the amount of diverted nuclear material increases in proportion to the size of the MUF uncertainty, and total amount of diverted nuclear material was assumed to be 8 kg, which corresponds to one significant quantity of plutonium. Virtual NMA system model (environment) and a divertor (agent) attempting to divert nuclear material were modeled to apply reinforcement learning. The agent is designed to receive a negative reward if an action attempting to divert is detected by the NMA system. Reinforcement learning automatically trains the agent to receive the maximum reward, and through this, the weakest diversion scenario can be derived.

As a result of the study, it was confirmed that the agent was trained to attempt to divert nuclear material in a direction with a low detection probability in this system model. Through these results, it is found that it was possible to sufficiently derive weak scenarios based on reinforcement learning. This technique considered in this study can suggest methods to derive and supplement weak diversion scenarios in NMA system in advance. However, in order to apply this technology smoothly, there are still issues to be solved, and further research will be needed in the future.

Acknowledgements

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Keywords: Nuclear material accountancy system, Material balance evaluation, Reinforcement learning, Diversion scenario, MUF uncertainty

Relationship Between Non-Compliance Notice and Corrective Order

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Regulators conduct inspections and issue non-compliance notice, and it is necessary to examine whether this is equivalent a corrective order. A corrective order imposes binding obligations on a business, and violations of it can result in sanctions such as fines, license revocation, or suspension. Therefore, if it is a corrective order, it must go through procedures such as prior notification under the Administrative Procedure Act. However, so far, no such procedures have been followed when issuing non-compliance notice. There are three possible interpretations of it: 1) the issuance of a non-compliance notice is not a corrective order but a part of an inspection, 2) the issuance of a non-compliance notice is a corrective order but not a disposition, and 3) the process of hearing opinions and prior notification was carried out during the inspection. However, if it is a part of an inspection, it should be issued by KINAC or KINS, which is entrusted with the inspection, but it is issued by the Nuclear Safety and Security Commission, and it is a disposition because it makes specific demands, and the corrective orders themselves have not gone through the procedure of hearing opinions. Therefore, in order for a non-compliance notice to be enforceable unlike a recommendation and to be issued by the Nuclear Safety and Security Commission instead of the inspection agency, the law should be amended and the procedure of prior notice and hearing opinions required by the Administrative Procedure Act should be carried out at the issuance stage.

Keywords: Non-compliance notice, Corrective order, Administrative procedure act

Analysis of the Xenon Isotopic Activity Ratios Considering Atmospheric Dispersion for the Scenarios of Xenon Releases After DPRK's Nuclear Test

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Understanding the dispersion of xenon isotopes following a nuclear test is critical for global security and falls within the remit of both the Comprehensive Nuclear-Test-Ban Treaty (CTBT) and the International Noble Gas Experiment (INGE). This paper aims to show if it is possible to discriminate the source of xenon releases based on the atmospheric dispersion of xenon isotopes using HYSPLIT.

Using ORIGEN and SERPENT simulations, four released scenarios are defined with four different fractionation times (i.e., 1 hour, 1 day, 10 days, and 30 days) after a 1kt TNT equivalent ^{235}U explosion event. These time-delayed release scenarios were selected to certify the possibility of mis-determining xenon release source.

We use the Lagrangian dispersion model for atmospheric dispersion to predict the concentration distribution of xenon isotopes under each scenario. The model allows us to better understand how these isotopes would distribute over time and space, offering valuable data for real-world detection efforts. To our knowledge, there have been no researches on the analysis of xenon isotopic ratios considering atmospheric dispersion. In this work, we focused on the atmospheric dispersion using HYSPLIT to characterize the xenon isotopic ratios from nuclear tests. In addition, we compared the xenon isotopic ratios obtained from the atmospheric dispersion with those from ORIGEN calculations, which would be helpful to discriminate the source of the xenon releases.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2103089).

Keywords: CTBT, IMS, Noble gas, Nuclear explosion, Atmospheric dispersion

Status of Anti-Neutrino Detection Technology for Reactor Monitoring in ROK

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Over the past decades, particle physics has made significant progress in characterizing neutrinos even if neutrinos have extremely small cross-section ($\sim 10^{-44}$ cm²), allowing them to penetrate any object. More recently, neutrino detection and analysis have indeed become valuable tools in various aspects of nuclear science and technology. Neutrinos are detected using various methods, including Inverse Beta Decay (IBD), Neutrino-electron scattering, and Coherent Neutrino-Nucleus Scattering (CNNS). For the detection of anti-neutrinos from nuclear reactor, the Inverse Beta Decay (IBD) is commonly considered with scintillators. Notable experiments in Korea, such as RENO and NEOS, have been conducted using the IBD method at the Hanbit Nuclear Power Plant since 2006. Additionally, the NEON experiment, which employs CNNS, which has a significantly larger reaction cross-section than IBD but its low-energy signal detection difficulty, has been ongoing since 2021. Based on the results of NEOS (2015-2020) the signal to noise is ~ 30 and IBD detection rate is ~ 2000 counts per day. The IBD event in nuclear power plants provides valuable information about reactor behavior. IBD count rates are in good agreement with the thermal power of the reactor. Furthermore, the neutrino energy spectrum can be used to estimate the fission isotope ratio of the reactor core, showing promise for obtaining reactor core information from anti-neutrino detection techniques. Neutrino detection in nuclear facilities provides valuable information about reactor behavior. However, as a surveillance technology neutrino detection faces challenges due to the very low cross-section, requiring efforts to overcome limitations related to detector size and signal acquisition time.

In 2008, the International Atomic Energy Agency (IAEA) included neutrino detection in its Research and Development (R&D) program for reactor safeguards. In January 2023, the IAEA organized a “Technical Meeting on Nuclear Data Needs for Antineutrino Spectra Applications” to discuss the latest developments and research results in this field. In summary, the use of neutrino detection in the nuclear field, particularly for reactor monitoring and safeguarding, has advanced significantly. Ongoing research and collaboration are expected to enhance our understanding of neutrinos and their applications in nuclear science and technology.

Keywords: Anti-neutrino, Inverse-beta-decay, Safeguards, Reactor monitoring

Security by Design for SMR Developments

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Most Small Modular Reactors currently under development are pursuing designs that can demonstrate flexibility in terms of construction and operation, and seeking to adopt innovative technologies to implement them, which is a very big challenge not only from the developer's perspective but also from the regulator's perspective. For the successful development of SMRs, it is necessary to move away from the existing prescriptive regulatory approach and exercise regulatory flexibility to sufficiently reflect design characteristics. The reason why SMR development is actively progressing around the world is to overcome the limitations of existing Light Water Reactors. Licensing advanced reactors such as MSR, VHTR, and microreactors requires a different approach from the existing conservative regulatory framework, taking into account changing regulatory environment. With the development of information technology and artificial intelligence, new types of threats are emerging, most of which are related to nuclear security.

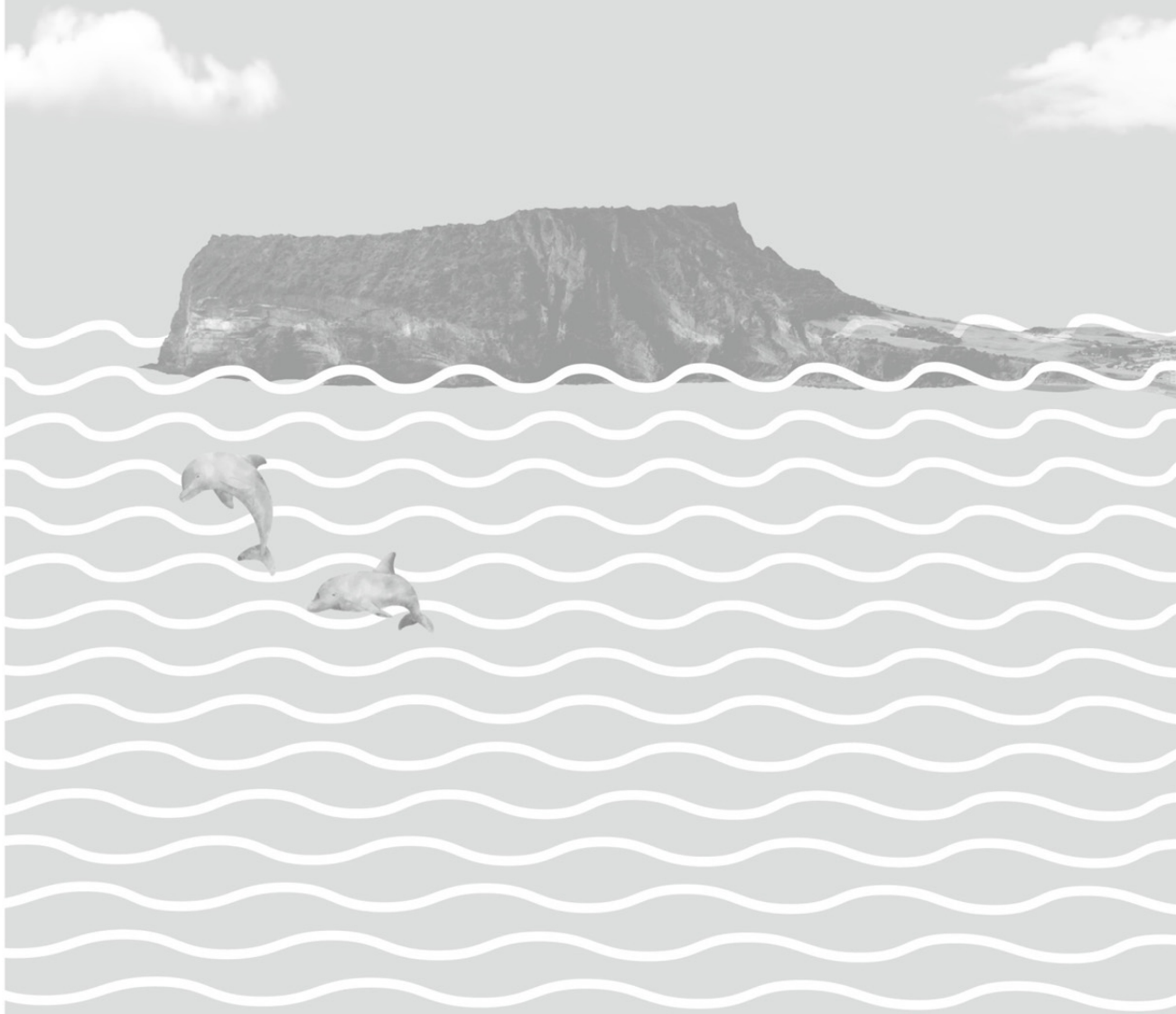
The IAEA, as well as leading countries such as the United States and Canada, require that safety, security, and safety measures be reflected in the early stages of design (Safety, Security, Safeguards by Design) and should be applied in the regulatory process. In addition, it is recommended to design a system that can achieve synergy effects by identifying in potential issues that may cause regulatory interference between safety and security (SSI, Safety-Security Interface).

The competitiveness of SMR in the international market will be highly dependent on the degree to reflect the importing countries' requirements. Since most SMRs currently under development do not have significant differences in safety goals, multi-purpose usability, etc., it is necessary to faithfully reflect the environmental factors necessary for SMR operation in the adopting country in the design. The major issues expected are mostly nuclear security-related rather than safety, and the physical characteristics of the site including the geopolitical conditions of the importing country must be taken into consideration. In this paper, the necessity of SMR nuclear security regulation and the way how to reflect the Security by Design on SMR will be presented.

Keywords: Nuclear security, Security by design, Safety-security interface, Small modular reactor, New types of threats

1분과

핵주기정책·규제 및 비확산 (POSTER)



A Study on Improving the National Safeguards Seals for the Wolsong Spent Fuel Dry Storage Facility

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A seal is one of the primary means of safeguards along with surveillance. The International Atomic Energy Agency (IAEA) uses various types of seals to verify the diversion of nuclear materials and is developing new seals according to the development of technology. Independent of the IAEA, ROK uses national safeguards seals for state-level regulation. A national safeguards inspector binds the nuclear material storage by combining a seal with a metal wire and checks the serial number of the RFID chip inserted in the seal with a reader.

The Wolsong spent fuel dry storage facility has 14 modules, each with 24 seals, and thus a maximum of 336 national seals will be installed. Although dependent on the sealing method, it takes about 5 minutes to verify one seal. As such, a considerable workforce is required for verification, and both the IAEA and the ROK are currently conducting random inspections. In addition, there are cases where verification is impossible because old seals are damaged due to harsh environments and long exchange periods.

Therefore, in this study, we analyzed cases in areas where sealing technology has been developed to improve the problems of the existing national safeguards seals. And we proposed a method for improving national seals by finding requirements of seals considering spent fuel dry storage facility characteristics.

In international logistics, sealing is essential in product transport verification, terrorism prevention, and tariff imposition. Accordingly, the field of container sealing has been extensively developed, and the International Organization for Standardization (ISO) has regulated the mechanical requirements of the seal as ISO 17712 and the electronic requirements as ISO 18185. Mechanical seals include metal and plastic seals and metal seals include bolt seals, ball seals, and cable seals. In addition, there are various electronic seals, such as radio frequency identification (RFID), near field communication (NFC), infrared (IR). Recently, there has been a trend to use active seals that have a built-in battery and can implement various additional functions.

Among the various seals, the main requirements for selecting seals suitable for dry storage facilities are as follows. First, use of a sealing tube longer than 10m should be possible. Second, it should have corrosion resistance so that it can be used for more than five years in the coastal area. Third, it must be a passive seal without a power supply. Fourth, it should not be overly costly. Finally, the seal verification time should be short. As a seal that satisfies these requirements, an electronic seal with application of the passive RFID method to the mechanical form of a metal cable seal is suitable. Since it is not an active seal, it is difficult to determine the time of breakage. Therefore, designing the seal such that the RFID is also damaged when the metal seal is broken will be helpful for verification.

In this study, the requirements for national safeguards seals in dry storage facilities were defined, and measures to improve the existing national seals were studied. Field applicability will be evaluated through future sealing device design and demonstration tests.

Keywords: Safeguards, Electronic seal, Dry storage facility

Nuclear Fuel Cycle-Related R&D Activities Classification Using Deep Learning

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This study aims to classify R&D activities related to the nuclear fuel cycle using the deep learning methodology. First, R&D data of the Republic of Korea were collected from the National Science & Technology Information Service (NTIS) for the years 2021, 2022, and 2023. We use keywords such as ‘nuclear,’ ‘uranium,’ ‘plutonium,’ and ‘thorium’ to find nuclear-related R&D projects in the NTIS database. Among the numerous R&D projects found through keyword searches, overlapping and medical-related R&D projects were excluded. Finally, 495 R&D projects conducted in 2021, 430 R&D projects conducted in 2022, and 296 R&D projects conducted in 2023 were obtained for analysis. After that, Safeguards experts determine whether the R&D projects are subject to declaration under the AP. The values of the content validity index (CVI) and content validity ratio (CVR) were used to verify whether the experts’ judgments were valid. The 1,218 collected and labeled data were then divided 8:2 into training and test datasets to see if deep learning could be applied to classify nuclear fuel cycle-related R&D activities. We use the Python and TensorFlow packages, including RNN, GRU, and CNN methods. First, the collected text information was preprocessed to remove punctuation marks and then tokenized to make it suitable for deep learning. After 20 epochs of training to classify the nuclear fuel cycle-related R&D activities, the RNN model achieved 97.30% accuracy and a 5.85% error rate on the validation dataset. The GRU model achieved 96.53% accuracy and a 9.06% error rate on the validation dataset. In comparison, the CNN model achieved 94.61% accuracy and a 2.57% error rate on the validation dataset. When applying the test dataset to each model, the RNN model had a test accuracy of 83.20%, the GRU test accuracy of 82.79%, and the CNN model had a test accuracy of 85.66% for the same dataset. This study applied deep learning models to labeled data judged by various experts, and the CNN model showed the best results. In the future, this study will continue to develop an optimum deep learning model that can classify nuclear fuel cycle-related R&D activities to achieve the purpose of safeguards measures from open-source data such as papers and articles.

Keywords: Nuclear fuel cycle, Text mining, Deep learning

A Study on Safeguards Approach for Uranium Extraction From Seawater

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Uranium extraction from seawater has been a topic of considerable interest over the past decades. However, Commercial facilities for uranium extraction from seawater have not yet been constructed due to its lack of economic feasibility. With the increasing demand for sustainable energy sources, there is a growing interest in eco-friendly uranium extraction methods. Despite this, the safeguards associated with these extraction techniques remain relatively under-researched, necessitating comprehensive studies that address both the economic feasibility and safeguards approach. The Korea Hydro & Nuclear Power Central Research Institute is poised to elucidate the economic value of uranium extraction from seawater and embark on research to extract Yellow Cake from seawater on a laboratory scale. Given these advancements, it becomes imperative to consider the approach to safeguards.

In this study, a comprehensive review was conducted to understand the relevant regulations that encompass both international obligations in partnership with the IAEA and domestic guidelines, specifically the Nuclear Safety Act. Emphasis was placed on a detailed examination of the IAEA's comprehensive safeguards agreement and its additional protocol, focusing on deriving the necessary regulatory timings, subjects, and methodologies for effective reporting and verification. We reviewed the safeguards guidelines and the IAEA policy to confirm the international non-proliferation obligations. The study also reviewed the impact of the State-Level Approach promoted by the IAEA and its implications on state-specific factors and evaluations of state technological advancement. Additionally, the regulatory aspects of extracted uranium as an internationally regulated material under the Nuclear Safety Act were critically assessed.

In conclusion, this study explains the international and domestic regulatory considerations for uranium extraction from seawater. Ultimately, this study will provide valuable understanding for policymakers, researchers, and practitioners involved in uranium extraction from seawater. Additionally, we expect that this study will contribute to establishing the safeguards approach and regulatory framework for the commercialization of uranium extraction from seawater in the ROK.

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Keywords: Uranium extraction, IAEA, Nuclear safety act, Safeguards, Regulatory considerations

A Proposal for the Revision of the Public Notice on the National Inspection of Nuclear Material Accounting and Control

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Among the public notices of the NSSC, five notices related to safeguards, including “Education of Nuclear Control, International Regulatory Materials, Preparation of Regulation of NMAC (Nuclear Material Accounting and Control), the National Inspection of NMAC, and Reporting of International Regulatory Materials” The regulations on the National Inspection of NMAC have remained the same since some revisions were made on December 26, 2017, raising the need to revise the public notice due to changes in the domestic and international safeguards regulatory environment. Accordingly, this paper analyzes the public notice of the National Inspection of NMAC and proposes the revision direction.

The regulation regarding the National Inspection of NMAC comprises sections such as Purpose and Definition, Types - Scope - Frequency of the National Inspection, Notification of the National Inspection’s plan, and Management of Violation. Appendices include the contents of the violation table, explanations regarding types of violations, and various forms related to the National Inspection, which are attached separately. IAEA mentioned that ROK was selected as a pilot country for the Improved SLA (State-Level Approach) project starting in November 2020. IAEA explained that a quantitative and standardized methodology was adopted and developed for this purpose. As a result, the Unannounced Inspection at LWR facilities will transition to the Random Interim Inspection. Additionally, the Physical Inventory Verification in CANDU facilities will increase to once a year per reactor. This status will change the frequency and intensity of inspection at domestic nuclear facilities. Furthermore, domestically, there is an ongoing trend of continuous growth and diversification of nuclear facilities. In light of the changing domestic and international safeguards environment, it is necessary to set a direction for revising the regulation regarding the National Inspection of NMAC that was partially amended in 2017 to align with the current status. Firstly, due to the increased burden on operators resulting from the increased number of IAEA inspections following the application of Improved SLA, there is a need to streamline the National Inspection of NMAC frequency to enhance overall regulatory efficiency. Furthermore, the definition section should also be revised to include matters related to the regulation to reflect the current reality accurately. Considering the operation and name changes of new domestic nuclear facilities, there may be a need to add or modify computer input codes.

While pursuing the revision of regulations regarding the National Inspection of NMAC, an analysis of the need for revision of other regulations related to safeguards should also be conducted, and directions should be set. Through this process, improving the regulatory framework that forms the basis of safeguards can help prevent confusion among operators and promote regulatory efficiency. We can better cope with these changes by proactively adapting to the rapidly changing domestic and international nuclear environment.

Keywords: Safeguards, Public notices, Regulations, National inspection

Improvement of Procedure of On-Site Check for Resident Inspectors in KINAC

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The KINAC resident inspectors are responsible for conducting on-site regulatory and intergovernmental support tasks related to safeguards, physical protection, and cybersecurity in each NSSC regional office. In nuclear material accounting and control, resident inspectors primarily perform tasks such as national inspections and technical support for IAEA inspections. However, with the increasing cases of non-compliance with the advance notification procedure by operators, there is a growing need for improvement in the role of resident inspectors in on-site regulation. In response to this situation, the safeguards division in KINAC has analyzed and improved the on-site check procedures of resident inspectors at LWR facilities.

The existing procedure outlines the process where resident inspectors receive the advance notification documents submitted by operators and utilize them as a reference for conducting weekly checks during the overhaul period when IAEA surveillance cameras are installed. Additionally, according to the attached forms specified in the procedure, resident inspectors are required to submit the check results report to the director of the safeguards division in KINAC every week and to the NSSC every month. The inspection items include checking the execution and changes of advance notification, verifying unnotified matters, discussing other issues, assessing the integrity of things such as the operational status of IAEA surveillance equipment, and so on. On April 13-14, 2023, the Safeguards division organized a two-day resident inspector's work-sharing workshop to discuss improvements in the on-site check procedures of resident inspectors at LWR facilities. During the workshop, a comparison and analysis were conducted between the existing procedures and actual on-site activities. Unnecessary tasks such as advance notification document reception and monthly reporting were eliminated, and the focus was shifted towards emphasizing essential tasks. The opinions of resident inspectors were taken into account to derive directions for improvement. The existing procedure was applicable only during Overhaul periods for resident inspectors. It has been improved by removing this limitation, allowing its use during routine times. Furthermore, the procedure has been enhanced by clarifying its purpose, scope, users, and definitions of terms and specifying responsibilities and authorities. Unnecessary terminology has been eliminated. Remarkably, the definition of advance notification has been detailed, and the reporting of check results has been simplified through weekly task reporting.

The Safeguards division in KINAC has strived to enhance the efficiency and simplification of on-site regulatory activities for resident inspectors at LWR facilities by improving their on-site check procedures. These improvement activities are expected to aid resident inspectors in effectively performing a wide range of tasks, including safeguards, physical protection, cybersecurity, and government support. In the future, it will be possible to continue refining the on-site check procedures by sharing the results of using the procedure in meetings and gathering various opinions from resident inspectors.

Keywords: Safeguards, Resident inspectors, On-site check, Procedures

ROK's Endeavors for IAEA Advanced Comprehensive Inspection Exercise at CANDU and LWR Facilities

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The Agency needs to maintain a solid and reliable foundation for recruited inspectors by providing practical training at commercial nuclear power plants. The Comprehensive Inspection Exercise (CIE) is a basic training which consists of a simulation of a Design Information Verification (DIV) Visit, a Physical Inventory Verification (PIV) at a nuclear power plant, including Complementary Access. The basic curriculum includes a pre-course session, auditing exercises, fresh fuel (bundles and assemblies) measurements, spent fuel (bundles and assemblies) measurements, verification of design features, as well as nuclear material flow. ROK has been holding the light-water reactor (LWR) / heavy-water reactor (CANDU) training course (CIE) from 2010 every year with about 2 weeks curriculum through MSSP (Member State Support Program). LWR and CANDU are operated by KHNP. To efficiently carry out the safeguards, IAEA receives the contribution through the ROK support program and implement R&D for the nuclear material inspection. ROK has been supporting and contributing total 22 tasks to IAEA in-cash and in-kind. Among them, this training provides for a course on safeguards verification activities at CANDU and LWR facilities. This course offers inspectors a unique opportunity to understand diversion scenarios and to familiarize themselves with instruments specifically used at CANDU and LWR facilities (OFPS and DCVD), as well as spent fuel dry storage transfer verification activities and dry storage dual sealing arrangements. KINAC performs PoC (Point of Contact) on behalf of NSSC and coordinates work between IAEA and KHNP. Additionally, KINAC first discusses with KHNP that can host light-water reactors and heavy-water reactors with KHNP at the beginning of each year. In order to hold a successful training, ROK plans and carries out a lot of process including agenda, accommodation, equipment movement, logistics and so on in consultation with the IAEA and facilities.

Acknowledgements

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Keywords: IAEA, LWR, CANDU, Training, MSSP, Safeguards

Evaluation of MUF Uncertainty Based on GUM Method for Benchmark Bulk Handling Facility

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The Republic of Korea (ROK), as a member state of the IAEA, is operating the State's System of Accounting for and Control (SSAC) and conducting independent national inspections. Furthermore, an evaluation methodology for the material unaccounted for (MUF) is being developed in ROK to enhance capabilities of national inspection.

Generally, physical and chemical changes of nuclear material are unavoidable due to the operating system and structure of facilities, an accumulation of material unaccounted for (MUF) has been issued. IAEA developed statistical MUF evaluation method that can be applied to all facilities around the world and it mainly focuses on the diversion detection of nuclear materials in facilities. However, in terms of the national safeguard inspection, an evaluation of accountancy in facilities is additionally needed. Therefore, in this research, a new approach to MUF evaluation is suggested, based on the Guide to the Expression of Uncertainty in Measurement (GUM) that an evaluation of measurement uncertainty factors is straightforward.

A hypothetical list of inventory items (LII) which has 6,118 items at the beginning and end of the material balance period, along with 360 inflow and outflow nuclear material items at a virtual fuel fabrication plant was employed for both the conventional IAEA MUF evaluation method and the proposed GUM-based method. To calculate the measurement uncertainty, it was assumed that an electronic balance, gravimetry, and a thermal ionization mass spectrometer were used for a measurement of the mass, concentration, and enrichment of ^{235}U , respectively. Additionally, it was considered that independent and correlated uncertainty factors were defined as random factors and systematic factors for the ease of uncertainty propagation by the GUM.

The total MUF uncertainties of IAEA (σ_{MUF}) and GUM (u_{MUF}) method were 37.951 and 36.692 kg, respectively, under the aforementioned assumptions. The difference is low, it was demonstrated that the GUM method is applicable to the MUF evaluation. The IAEA method demonstrated its applicability to all nuclear facilities, but its calculated errors exhibited low traceability due to its simplification. In contrast, the calculated uncertainty based on the GUM method exhibited high reliability and traceability, as it allows for individual management of measurement uncertainty based on the facility's accounting information.

Consequently, the application of the GUM approach could offer more benefits than the conventional IAEA method in cases of national safeguard inspections where factor analysis is required for MUF assessment.

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Keywords: Nuclear safeguards, Material unaccounted for evaluation, National inspection, Guide to the expression of uncertainty in measurement, Benchmark case evaluation

The Actual Application on the Export Control of ITT at KAERI: Focused on the Implementation of Inner Self-Audit on the Organization of ICP

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In compliance with the amended export control of strategic items and technology in Jan. 2014, KAERI should pay attention to the export control of ITT (Intangible Technology Transfer). To control an ITT (Intangible Technologies Transfer) effectively and efficiently, the Korean government encourages the R&D institute and universities obtaining the ICP (Internal Compliance Program) from the relevant authority, MOTIE. This means that the exporters can control the ITT by themselves, because the exporters know very well the counterparts of the trading and the exporting items and technologies. In fact, ICP is for export control of dual-use items and technology in Korea. However, KAERI has tried to obtain a license from the authority, MOTIE. In an effort to do so, KAERI completed enacting a new internal self-regulation for export controls in 2016, and proceeded to apply for an 'AA' license of ICP in 2017 and obtained the ICP license in 2018 and re-obtained the license in 2021 from the MOTIE. In light of KAERI's case, to obtain the 'AA' license of ICP is one of the best methods to increase the ability of export controls. As of now, there is no R&D institutes sponsored by the Korean government to obtain the 'AA' license of ICP except KAERI. KAERI can provide the actual methods as a standard case to the R&D institutes in Korea for obtaining an 'AA' license of ICP.

According to the internal regulation of KAERI for export control, KAERI implemented an inner self-audit for export control in Nov. 2022. This is the first real self-audit for export control at KAERI. The main purpose of the self-audit is to check the transfer management of ITT and the relationship of relevant office through the interview of the staffs in the ICP organization. KAERI self-audit planed specifically and implemented for the achievement of the basic principal of self-audit. The specific contents of this self-audit is as follows

- The interview of the relevant offices: physical protection office, manpower planning office, manpower management office, nuclear education and training center, technology transfer office and international cooperation office, nuclear control and management office
- Building the self-audit checklist considering the characteristics of each office
- The confirmation of the inner procedure and the status of management on the export controls

Through the interview of the relevant office, KAERI checked the inner procedure and the status of management on the export controls and tried to provide the supplementary measures of each relevant offices. The followings are the main results of the inner self-audit implemented in Nov. 2022.

- Generally, the staffs know the meaning and relevant regulation such as foreigner's management and the intangible technology transfer
- Each office reflects the necessities of export controls on the relevant regulation and procedures and make DB for the proper duty. However, there is no indication for export controls on the DB
- In the case of foreigner's temporary visit for simple work and site tour, there is a difficult situation not to be able to check all the visitors by checking the denial lists
- If necessary, KAERI may build the TFT (Task force Team) for the efficiency of export controls
- Others

Keywords: Export control of ITT, Inner self-audit system for export control, Internal compliance program, a level 'AA' license of ICP, The amended export control of strategic items and technology

Cases of Overseas Drone Threats at Nuclear Power Plants

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As drones expand beyond military purposes to the private sector, the level of use of drones in various fields is increasing. However, the world was shocked by the attempt to attack with a drone equipped with a C4 bomb in the US and the attempt to assassinate a head of state using a drone in Venezuela. Drone threats to domestic nuclear power plants are also increasing due to the expansion of drone use, terrorist threats, and North Korea's invasion of drones.

Overseas, various drone threats to nuclear power plants have occurred. In October 2014, French electricity company Electricite de France confirmed that it had observed unauthorized drones over seven nuclear power plants across France. A drone threat occurred at the Savannah River Site (SRS), a U.S. Department of Energy facility that processes and stores nuclear materials. In 2016, eight drones were observed by security personnel. In 2016, a drone flew over the cooling tower of the Liebstadt nuclear power plant in Switzerland, and publicly shared the filmed video on YouTube. In July 2018, Greenpeace activists intentionally crashed a drone into the outer wall of the spent fuel building in Boughey, France. In January 2019, they used drones to drop smoke bombs and release videos at Orangeto's nuclear facility containing irradiated fuel. In January 22, Sweden saw drones flying over three nuclear power plants. Drone was also seen at the Forsmark nuclear power plant on Friday and at two other Swedish nuclear power plants in Oskarshamn and Ringhals on Monday.

Anti-drone technology to counter the threat of drone terrorism is also developing. Anti-drone technology detects, tracks, and identifies illegal drones to neutralize them. Various technologies such as radar, EO/IR cameras, Lidar, sensor, and RF scanners are being developed for drone detection. Depending on the detection technology, it has advantages such as detection distance and remote control drone detection. However, there are also disadvantages, such as obstacles, weather condition, and the inability to detect drones that do not transmit signals. Methods such as jammer, capture, and destruction have been developed for incapacitation technology. While it has advantages such as stability and portability, it has disadvantages such as limited use and damage to the surroundings.

Accordingly, it is necessary to draw realistic measures to defend against the threat of nuclear power plants by paying constant attention to the various detection, identification, and neutralization anti-drone systems that continue to evolve.

Keywords: Drone threat, Anti-drone, Security threat

An Analysis of DPRK's WMD Programs and Sanctions Evasion Activities

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The Democratic People's Republic of Korea (DPRK) has been exporting weapons of mass destruction (WMD) to the volatile Middle East and Africa. It is expecting that military illicit activities would isolate DPRK economically, as it has been placed on multiple sanctions lists, including UN sanctions, multilateral export control regime sanctions, and country-specific sanctions. However, DPRK funds its WMD programs through various sanctions evasion activities.

DPRK's primary sanctions evasion activities include obtaining foreign currency, acquiring dual-use or restricted technology, smuggling, and money laundering, which are global in scope. This study analyzes the sanctions evasion activities used by DPRK to acquire economic, material, and technological resources for its WMD program and devises ways to disrupt these evasions effectively.

First, the international community should strengthen export controls by encouraging states with weak export control regimes to join international organizations and conventions to limit DPRK arms and technology exports. Second, states should improve their intelligence gathering and analysis capabilities by sharing information on DPRK's evasion activities and working together. This will help identify and counter DPRK's evasion techniques and networks. Third, the international community should strengthen cooperation on DPRK's evasion efforts. This can be done by strengthening cooperation with states and entities that enforce international sanctions and by working with relevant agencies such as customs, immigration, and police to track and interdict the movement of funds and assets used for evasion. Fourth, publicize DPRK's illicit activities and apply diplomatic pressure. Diplomatic pressure can lead to more states and entities to enforce sanctions.

In conclusion, these strategies are expected to deter DPRK's illicit activities; but to sanction DPRK effectively, it is essential to continue to adjust and refine the strategies in response to DPRK's evolving sanctions evasion efforts. The results of this research are expected to prevent WMD proliferation through DPRK by blocking or reducing the risk of sanctions evasion.

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Keywords: DPRK, WMD programs, Sanctions evasion activities

Analysis of Nuclear Export Control Trends and Related Organizations’ Outreach Activities

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The Korea Institute of Nuclear Nonproliferation and Control (KINAC) conducts outreach to promote and educate regulated entities on the export control regime’s purpose, importance, and implementation. Outreach activities help to reduce regulatory blind spots and minimize domestic and international penalties for non-compliance. The need for outreach is growing as domestic and international policies are changing rapidly, and the scope of export regulations is expanding due to increased exports of nuclear power plants. In order to explore the long-term development direction of outreach activities, we will analyze the trends of nuclear export control and the outreach activities of related organizations.

Here are some key trends in nuclear export controls. In recent years, countries worldwide have been reorganizing their supply chains for critical industries, focusing on their own and friendly countries, and strengthening their trade policies in security aspects such as export control and technology protection. Following the trend of international sanctions against Russia, the Korean government has implemented domestic export control measures similar to those of the international community, such as blocking the export of strategic goods to Russia. In addition, the number of strategic goods classifications and export licenses has been increasing as Korea promotes the export of new nuclear power plants. In line with carbon neutrality, it is expected to revitalize and diversify nuclear energy-related export businesses, such as joint research on fourth-generation nuclear power plants and SMRs. Finally, the scope of exports is expanding from ‘goods’ such as existing nuclear reactors to ‘technology’-oriented transfers. The means of technology transfer are diversifying with the development of information and communication technologies such as cloud services, email, video conferencing, and large-capacity removable storage devices.

Next, look at the outreach activities of nuclear export control organizations. The Korean Security Agency of Trade and Industry (KOSTI) is an organization that implements export controls on dual-use items. It puts much effort into one-on-one consulting services with companies and has established and operated various online training programs. It also actively utilizes online promotional materials such as card news and videos. The export control agencies of major countries have a common trend of expanding outreach to research institutions, providing export control guides tailored to the characteristics of each field, holding annual seminars and conferences, and operating educational programs.

Keywords: Nuclear, Export control, Outreach, Trends, KINAC

A Case Study of Domestic/International Drone Threat/Terror and Classification of Drone Threat Types

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In recent times, drone technology has been rapidly advancing and becoming increasingly popular. Furthermore, there has been an increase in the number of crimes and terrorism cases targeting national facilities using drones. This study aims to categorize the types of drone threats that could pose future risks to nuclear power plants. For this study, we are investigating domestic and foreign drone terrorism cases and identifying the specifications of drones used. It has been confirmed that products from Chinese DJI companies have frequently appeared as commercial drones used in terrorism. This suggests that conversion of general commercial drones into weapons can be effectively utilized for terrorist activities. There is an elevated risk of terrorism involving multiple small drones. Nuclear power plants must also devise protective measures against a large influx of drones. Additionally, it is predicted that North Korea is developing drones equipped with return technology through GPS-based autonomous mission flights. North Korea's drones are presumed to have been converted from Chinese drones (SKY-09P, UV10CAM, etc.). According to the analysis based on the weight and size of the drone, drones weighting less than 150 kg and wingspan of less than 3 m are used for terrorism. To effectively detect drones, it is necessary to implement measures such as integrating and deploying various equipment to compensate for equipment limitations (radio waves, radar, video, sound, etc.). In the case of long-distance flight, a number of fixed-wing drones capable of autonomous mission flight and long-distance flight were used. North Korea's drones use GPS-based autonomous mission flights, so it is necessary to prepare drones that do not transmit RF signals to detect them. Both RF signal detection and GPS jamming should be carried out, with GPS jamming taking precedence, even in the case of fixed-wing drones. The results of this study could contribute to enhancing the level of physical protection of nuclear power plants.

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Keywords: Drone, Threat, Terror, Detection, Neutralization

Review on Development and Status of Nuclear Cooperation Agreement Implement Module in NEPS

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The ROK government has developed the Nuclear Export and Control System (NEPS) to implement export control activities. Although it was launched in 2008 as a system that can work with classification, licensing, nuclear material approval, government-to-government assurance, complying with nuclear cooperation agreement (NCA) handled through official documents. In order to enhance systematic management for items subject to NCA, KINAC developed a new module for the procedure (hereinafter referred to as “NCA module”) and opened it in 2022. This paper presents the module’s development background, key features, and current operation status.

The NCA module prioritizes functional expansion and flexibility, distinct from other tasks for the following reasons. First, the export control duties of classification, export license, and approval for NM are based on domestic law, leading to predetermined target items, application forms, and processes that change only through statutory amendments. In contrast, the implementation of NCA has numerous procedural variables, varying across countries in scope, content, and procedures. Therefore, if the function is over-standardized, there would be many exceptions that the system cannot resolve in practice. Second, the existing NEPS process entails a one-time decision or approval for each application, while the implementation of the agreement encompasses four related procedures for each item: prior notification, written confirmation, shipment notification, and receipt confirmation. Even some steps may be omitted depending on the case. The other difference is the working process. The implementation of NCA must be initiated from the government, so the existing methods, beginning with the licensee filling a form, cannot be adopted as it is.

The NCA module has adopted a new reference numbering system to resolve these challenges. It enables the creation of multiple procedures under one reference number on an item to expand the tasks and make it possible to omit some steps or to reflect case-by-case concerns in each stage. It also provides a consolidated view of multiple notifications related to a single item, ensuring to deal with even long-running tasks without missing any obligations until the final procedure. Moreover, some of the data in the NCA module is extensible by allowing users to manage the list themselves. For example, the system can respond to new agreements by allowing users to add and modify codes that distinguish counterparty countries.

As a result, the current NCA module accommodates a variety of implementation scenarios, including split shipments, the procedural omissions, and the modification of additional counterparties, offering enhanced flexibility and adaptability.

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Keywords: Export control, Nuclear cooperation agreement module, System development, Procedural flexibility

NEPS-OTS Integration Methodology for One-Stop Import and Export Control for Internationally Controlled Materials

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Korea has signed nuclear cooperation agreements (NCA) with 29 countries. Nuclear materials, materials, equipment, and technology transferred under the agreements are “internationally controlled materials (ICM)” under the Nuclear Safety Act. The main obligations imposed on those items include ensuring peaceful use, safeguards, physical protection, annual inventory reporting, and retransfer with supplier prior consent. The Nuclear Export and Import Control System (NEPS) handles the export control procedures for transferring ICM. After import, inventory management for ICM in Korea would be transitioning to an item-based system through the Obligation Tracking System for internationally controlled item (OTS) currently under development. A one-stop import and export control system for ICM can be established when information is well-linked between these two systems. This paper aims to derive a methodology for integrating NEPS and OTS.

NEPS-OTS coupling begins at the receipt confirmation and shipment notification stages in NEPS. When importing ICM under NCA, the inventory change (code RF: receipt foreign) is entered in OTS by getting the information that has completed the receipt confirmation in NEPS. Conversely, during export, the information that has completed the shipment notification procedure in NEPS is linked to the OTS so that the entire cycle from import to re-export of the ICM can be concluded. Inventory verification for retransfer, checking that the book inventory remains positive value, is impossible under the current system. This issue can be resolved by enabling inventory information in OTS to be displayed in NEPS.

Determining when and how to generate the obligation code for imported ICM is essential for NEPS-OTS coupling. Manual input may be necessary for some cases with multiple obligations. Nevertheless, it is more efficient from a system communication protocol to automatically generate and impose a single obligation based on the supplier country information in NEPS. Moreover, it is important to automatically link crucial information available in NEPS to reduce the administrative burden on OTS users and discrepancies between systems. Most required OTS data, such as country obligation, item categories, quantity, physical or chemical form, and receipt date, can be directly linked from NEPS. However, NEPS improvement is needed for digitizing the receiver information and facility data, like the material balance area.

The NEPS-OTS integration involves sharing data as a system and encompasses the connection between export control and inventory management. Future work to link some information in NEPS-OTS with the KSIS could be suggested to enhance efficiency and effectiveness in managing ICM.

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Keywords: Export control, Nuclear cooperation agreement (NCA), Internationally controlled materials, System integration

International Case Studies of Compliance Program (CP) on Nuclear Export Control

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Strategic item export control aims to maintain international peace and safety and serves as a significant nuclear non-proliferation regime that directly impacts a nation's security. Therefore, establishing an autonomous export control system at the state level is crucial, and one of the most efficient methods to achieve this is by enhancing an export company's management system. Accordingly, many advanced countries, such as the United States, Europe, and Japan, have operated their own internal compliance programs (CP or ICP) to manage and screen the export of strategic items as a corporate social responsibility and risk mitigation measure.

In Korea, which has a high dependence on trade, the need for CP was continuously confirmed, but the system was introduced in 2004, relatively late compared to other advanced countries. So far, the Korean government has made steady efforts to develop and establish the system and is actively encouraging businesses to obtain Compliance Program certification to autonomously manage strategic items.

Major technologically advanced countries utilize technology transfer as a tool for economic sanctions, trade security, and strategic technology management, and they continue to strengthen their control regimes. In these countries, CP certification is considered a standard practice for export control among mid-sized and large enterprises. It serves as a vital risk management system that protects companies from unforeseen incidents. However, in Korea, the application of CP under the Foreign Trade Act is limited to dual-use items and does not extend to the nuclear export control system.

Therefore, this paper analyzes international cases and CP requirements in countries like the United States, Japan, Europe, and Singapore. As a result of the review, the application of CP into Korea's nuclear export control could be a coexistence means that can strengthen supply chain control as well as provide benefits not to impede technical research, international trade, and exchanges.

Keywords: Export control, Compliance program (CP), Non-proliferation, Trigger list, Strategic item

Evaluation of Nuclear Proliferation Risk Through Nuclear Material Attractiveness in Nuclear Fuel Cycle

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As the demand for nuclear power increases as a means to achieve carbon neutrality, concerns about nuclear proliferation have also grown. Consequently, significant researches have conducted to enhance nuclear non-proliferation resistance. Among these research, nuclear material attractiveness is a methodology used to evaluate how appealing a particular material is for potential use in nuclear weapons, based on the characteristics of that material. Existing nuclear material attractiveness assessments focused on materials like U, Pu, and TRU, which could be directly used in the production of nuclear weapons. However, these assessments did not consider how the properties of nuclear materials change throughout the nuclear fuel cycle, with each facility process.

This study assumed a scenario of the nuclear fuel cycle of graphite reduction reactors and analyzed including enrichment facilities and PUREX. This study used the FOM (Figure-Of-Merit) method developed by LANL (Los Alamos National Laboratory) for evaluating the nuclear material attractiveness. The FOM formula consists of three parameters such as critical mass, heat content, and dose. The critical mass of target materials and the dose evaluation were conducted using the Monte Carlo N-Particle code. The heat content was calculated using the ORIGEN code embedded in the SCALE code. In particular, if U-238 is dominant in the facility's materials, such as mining and refining facilities, and critical mass evaluation is impractical. Therefore, 1SQ (Significant Quantity) of that uranium was assumed as the critical mass value for the FOM evaluation, even though 1SQ is not identical to the critical mass.

As a result of this study, the attractiveness of Pu produced by PUREX among all nuclear fuel cycle facilities was 2.7616, which was the most attractive to be diverted to nuclear weapons. Through this study, it was shown that the proliferation risk of the nuclear facilities in the nuclear fuel cycle and risk of diversion among those facilities.

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Keywords: Non-proliferation, Nuclear material attractiveness, Nuclear fuel cycle, FOM

A Review of U.S. Legislation on Unauthorized Technology Transfer

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In the contemporary digital age, the rapid pace of technological advancement has elevated concerns regarding unauthorized technology transfers. These illicit transfers not only pose economic threats but also have the potential to compromise national security, strain international relationships, and impede technological innovation. Recognizing these challenges, the United States, as a technological leader, has implemented stringent laws and regulations to counteract such activities.

Unauthorized technology transfer or disclosure is treated seriously within the United States. It can be subject to various laws, including export control regulations, economic sanctions, and Non-proliferation laws. Export control measures such as the International Traffic in Arms Regulations (ITAR), targeting defense items, and the Export Administration Regulations (EAR), focused on dual-use items, are enforced. The Office of Foreign Assets Control (OFAC) regulations oversee U.S. trade sanctions. At the same time, the Iran, North Korea, and Syria Non-proliferation Act (INKSNA) penalizes those aiding proliferation activities in these countries. Such unauthorized transfers could undermine global security by violating international agreements like the Nuclear Non-Proliferation Treaty. The United States employs a multi-faceted approach to counter these threats, including international partnerships, strengthened export controls, diplomatic efforts, and rigorous enforcement.

Beyond the legal aspects, unauthorized technology transfer carries significant economic, national security, and global trade implications. Intellectual property (IP) theft can result in economic losses, national security risks, and strained trade relations. Legal protections against IP theft encompass Patents, Trade Secrets, Copyright, and Trademark Acts.

In conclusion, unauthorized technology transfers and IP theft present multi-faceted challenges with far-reaching implications for global security, economic prosperity, and international relations. The comprehensive approach taken by the United States, which combines legal measures, diplomatic engagement, and collaborative efforts, serves as a valuable example that South Korea can learn from and enhance to keep technological advancements and ensure a secure digital future.

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Keywords: Technology transfer, National security, Export control, Intellect property, United states

Emerging Technologies and Export Controls in the Nuclear Energy Sector

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Emerging technologies are innovative technologies currently under development or in the early stages of introduction. These technologies have the potential to impact a wide range of industries and sectors significantly and may, therefore, be subject to export controls. The list of emerging technologies subject to export controls varies from country to country and constantly changes as new technologies are developed.

For example, the U.S., EU, and South Korea have responded to these changes by adding software and technologies related to artificial intelligence and machine learning to their export control lists. Nevertheless, export control of emerging technologies still presents challenges and limitations. The rapid pace of technological advancement makes it difficult for export control regulations to keep up. For export control purposes, international cooperation on information sharing and control methods is necessary for most countries to control similar items.

Several new technologies in the nuclear field may be subject to export controls. These technologies include advanced reactors, nuclear fuel cycle technologies, and nuclear waste management technologies. Small modular reactors (SMRs) and fourth-generation reactors are being developed as advanced technologies, and new technologies are being developed to improve the nuclear fuel cycle. There is also active development of technologies for space applications utilizing nuclear reactors, such as the Nuclear Thermal Propulsion System and the Nuclear Electric Propulsion System.

As these technologies may include new systems and items not in existing export control, they may pose a proliferation risk or may include software design know-how for advanced materials, it is necessary to consider whether and how they should be subject to export control to prevent nuclear proliferation.

Overall, export controls are an essential issue in the emerging technology and nuclear energy sectors. Countries are moving toward strengthening regulations and international cooperation to overcome these challenges and ensure safe technology transfer, and South Korea should actively participate and lead this trend.

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Keywords: Emerging technology, Export controls, Advanced reactors, Space application

Project Management Standardization for R&D Quality Improvements in KINAC

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ISO 9001:2005 is the international standard for implementing a Quality Management System (QMS), which provides a framework and principles for managing an organization's quality management. The aim is to ensure that the organization continuously provides products and services that satisfy regulatory requirements. The "process approach" in ISO 9001 is defined as a systematic method of achieving organizational goals by comprehending and managing the interconnected processes as a cohesive system. Recently, KINAC has decided to develop standard processes in the field of R&D and performance management based on the framework of the ISO 9001:2015 quality management system. The objective of this study is to establish standardized processes for conducting research and development, as well as managing the outputs and performance of R&D activities. It involves identifying, designing, implementing, monitoring, and continually improving processes to ensure consistency, efficiency, and effective management of KINAC R&D and its achievements. Firstly, R&D and the research performance management process were defined, and the processes were categorized by function according to the requirements of ISO 9001:2015. Second, the ISO 9001 requirements were compared to the institute's existing regulations and documents in order to identify any additional processes and procedures needed to meet the quality management requirements. Finally, the lists of quality documentation were determined for the institute's QMS. As a result, a total of 30 QMS documents were listed, including 1 manual, 12 quality processes and procedures, and 17 quality instructions. The documents can be categorized into four process groups: the management and planning process group, the R&D and achievements management process group, the analysis and improvement process group, and the support process group. All input and output information of each process are connected and interrelated. The implementation of quality management standards and procedures for R&D in KINAC could lead to improved research practices, more reliable data collection and analysis, and increased efficiency in conducting R&D activities. For further study, it is planned to create detailed, high-quality documents that adhere to standard requirements and guidelines.

Keywords: Quality management system, Processes approach, R&D, Performance management process, ISO 9001:2015

A Study of Qualitative Methods for R&D Projects Based on the Standards and Guidelines for Project Management

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The National R&D Innovation Act emphasizes the improvement of the quality of R&D activities. The research institute is making efforts to improve the quality of research and effectively manage research implementation. KINAC has conducted various R&D projects regarding nuclear non-proliferation and nuclear security, and their scope and scale have been gradually more widened and increased. It consequently becomes important how to successfully manage research projects and ensure their qualification with the growth and complexity of research in KINAC. Unfortunately, no attempt was made to introduce and apply project management methodologies. Therefore, the objective of this study is to introduce project management standards and guidelines as an initial step towards improving the overall research quality of the institute. Project management is the well-organized application of knowledge and techniques to efficiently and effectively initiate, plan, control, and close projects, in order to achieve specific goals and meet success criteria. There are some guidelines regarding project management, including PMBOK (the Project Management Body of Knowledge), PRINCE2 (Projects in Controlled Environments), ISO 21500 (Guidance on Project Management), and PMP (Project Management Professionals), etc. They are international standards that consist of processes, guidelines, and best practices for project management. They provide structured processes and approaches to plan, execute, monitor, control, and complete projects. By reviewing the guidelines, the commonly important factors, including schedule, cost, quality, resources, communication, and risk management were introduced to apply to KINAC R&D project implementation. In addition to the management standards, systematic efforts are also continued to enhance the R&D qualities of the institute. These efforts include the implementation of a quality management system (ISO 9001:2015), development of an integrated research achievements management system, regulation development, and distribution of guidebooks for project managers and researchers. These efforts have been evaluated as improving the quality of the research.

Keywords: Project management standards and guidelines, PMBOK, PRINCE2, ISO 21500, PMP

Analysis of KINAC's Nuclear Export Control Outreach Programs

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The Korea Institute of Nuclear Nonproliferation and Control (KINAC) conducts various outreach activities, such as publishing brochures and holding seminars and briefings, to make regulated parties aware of the importance and necessity of the export control regime. The outreach program aims to increase compliance rates by generating interest in the export control regime among recipients and to increase communication to support compliance. In order to explore the long-term development of outreach activities, we analyze how KINAC conducts outreach.

KINAC conducts nuclear export control outreach to organizations that deal with trigger list items and related technologies. Educational institutions with nuclear energy-related departments, research institutes related to nuclear energy and materials, and industrial companies that handle equipment used in nuclear power plants or nuclear materials were selected for outreach. The outreach program provides information on the export control regime for trigger list items, the strategic technology control regime, and the Nuclear Cooperation Agreement.

KINAC's outreach programs can be categorized into education, exhibition, and publication. In the education program, we hold workshops and seminars for industrial companies, with customized content that considers the items handled by companies and the nature of technology transfer. We provide training for educational and research institutions focused on conducting research tasks and projects and transferring technology accordingly. As a result of the education program, there is a regret that the education for SMEs and educational institutions is not directly linked to the implementation of nuclear export control.

The exhibition program operated a booth at nuclear-related exhibitions at least once a year. The booth distributed brochures or publications on the export control regime, conducted surveys to investigate awareness of the regime and conducted on-site consultations. The exhibition program effectively increased the understanding of the export control regime among the general public and potential regulated parties. However, it was only sometimes linked to the actual implementation of nuclear export control.

The publication program produced promotional materials for use at education and exhibitions, as well as guidance materials on new and revised regulations. It used the agency's online media to provide information on new and revised export control legislation and related issues. As a result of the publication program, various existing publications explaining the export control regime were consolidated into a single publication, increasing the efficiency and satisfaction of outreach.

Keywords: Nuclear, Export control, Outreach, Program, KINAC

Investigation and Analysis of AI Language Models for the Development of the Nuclear Export and Import Control Associated Search System

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The Nuclear Export and Import Control System (NEPS) is currently in operation for nuclear export and import control. To ensure consistent and efficient control, various computational systems are either already in place or being developed. With numerous scattered systems, it becomes crucial to integrate the databases from each to maximize their utility. In order to effectively utilize these scattered computer systems, it is necessary to integrate the databases of each system and develop an associated search system that can be used for integrated databases, so we investigated and analyzed the AI language model that can be applied to the associated search system.

Language Models (LM) are primarily divided into two categories: understanding and generative. Understanding Language Models aim to precisely comprehend and analyze the provided text's meaning. They consider the text's bidirectional context to understand its deeper implications and are used in tasks such as text classification, sentiment analysis, question answering, and named entity recognition. In contrast, Generative Language Models focus on generating new text based on the given context. They produce new textual content continuously and are beneficial for text generation, machine translation, sentence completion, and storytelling. Given that the primary purpose of our associated search system is to comprehend user sentences or queries accurately, understanding language models are deemed more suitable.

Among the understanding language models, we examined BERT and its derivatives, RoBERTa and DeBERTa. BERT (Bidirectional Encoder Representations from Transformers) uses a Bidirectional Transformer Encoder to understand the sentence context and engages in pre-training by predicting 'MASKED' segments. RoBERTa (A Robustly Optimized BERT Pre-training Approach) enhances BERT by optimizing its training methods and data processing. Although its core architecture is similar to BERT, it incorporates improvements such as eliminating the NSP (Next Sentence Prediction) task, introducing dynamic masking techniques, and refining training data volume, methodologies, and hyperparameters. DeBERTa (Decoding-enhanced BERT with disentangled attention) introduces a disentangled attention mechanism to the BERT architecture, calculating the relative importance score between word pairs to distribute attention more effectively and improve performance.

In analyzing the three models, RoBERTa and DeBERTa demonstrated superior performance compared to BERT. However, considering factors like the acquisition and processing of training data, training time, and associated costs, these superior models may require additional efforts and resources. It's therefore crucial to select a language model by evaluating the economic implications, objectives, training strategies, performance-assessing datasets, and hardware environments. Additionally, it was noted that by fine-tuning with methods from RoBERTa or DeBERTa based on pre-trained BERT models, the training speed could be significantly improved.

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Keywords: Nuclear export, Nuclear import, Associated search system, Language model, NEPS

Considerations for Application an Internal Compliance Program (ICP) for Trigger List Items

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An Internal Compliance Program (ICP) is a system through which enterprise internally manage their own export control processes to ensure compliance with domestic export control laws. Around the world, ICPs are actively utilized as a means of export control for strategic items. However, they are not mostly applied to the Trigger List Items. However, advanced countries such as the United States and the Nuclear Suppliers Group (NSG) have been actively researching the potential application of ICPs to the Trigger List Items recently. This paper suggests additional considerations that should be taken into account when applying an ICP to the Trigger List Items.

The key elements of classical ICP include Top-level management commitment to compliance; Risk analysis; Organizational structure/chain of responsibilities; Human and technical resources allocated to the management of exports; Workflow management and operational procedures; Record-keeping and documentation; Selection of staff; training and awareness-raising; Process-/System-related controls (ICP audit)/Corrective Measures; Physical and technical security.

An ICP for Trigger List Items must encompass all these core elements. Additionally, as the nuclear industry often involves collaborative projects participating with various companies, the effectiveness of the ICP could be enhanced through the operation of consultation groups among participating companies. Furthermore, enterprises must take into account the unique characteristics of Trigger List Items that differ from other strategic items, when making requirements of the ICP establishment.

First, export requirements related to safety measures and physical protection should be reviewed to export the Trigger List Items. The procedure and obligations in aspects of internationally controlled items should also be reviewed. Moreover, active support from enterprises for GTGA procedures should also be included, since the Government to Government Assurance (GTGA) procedure is additionally required for the export of Trigger List Items, in contrast to other strategic items. Additionally, for materials categorized within Trigger List Items, such as deuterium and heavy water, should be controlled based on their end-use and cumulative quantity, which Government cannot effectively manage without enterprise supports. Therefore, enterprises must establish an internal material management system based on the end-use and cumulative quantity of these materials under ICP.

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Keywords: Trigger list items, Export controls, Internal compliance program (ICP), Strategic items

The Study on Institutional Improvements for Export Control the Items Subject to the Bilateral Nuclear Cooperation Agreements

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When exporting nuclear-related items, export control is required from two perspectives: the control of “Trigger List Items” as controlled by Nuclear Supplier Groups (NSG) and the control of the “Items Subject to the Agreement” as specified in bilateral Nuclear Cooperation Agreements. While Trigger List Items and Items Subject to the Agreement are largely similar, there are some items where they do not overlap. Furthermore, national law for controlling each item is different. The Trigger List Items are governed by the Foreign Trade Act, and the Items Subject to the Agreement (Internationally Controlled Items) are governed by the Nuclear Safety Act. As a result, the detailed procedures and requirements for controlling each item are quite distinct.

For the Trigger List Items, export license must be obtained in accordance with the Foreign Trade Act. The details such as responsible authority, the items subject to license, license requirements and procedures, penalties are specified in the Public Notice on Import and Export of Strategic Goods. For the Items Subject to the Agreement, the process and obligations set forth in bilateral agreements and related administrative agreements are fulfilled in accordance with the Nuclear Safety Act. However, in contrast to the Trigger List Items, the details for complying with the agreements are not specified legally.

Since most of the Items Subject to the Agreement are fall within the category of the Trigger List Items, the obligations in accordance with the agreements are reviewed and implemented during the export license assessment process. However, if the Items Subject to the Agreement are not are fall within the category of the Trigger List Items, there is a risk of control omission. For example, this applies to cases of exporting tritium and tritium removal facilities, which are not the Trigger List Items, to Canada and Romania. Moreover, since subjects to the agreement and compliance procedures are respectively different for 29 bilateral Nuclear Cooperation Agreements signed with different countries, it is difficult for enterprise to recognize the appropriate procedures and obligations under the agreement by their own.

The bilateral Nuclear Cooperation Agreements establish legal obligations between state parties while NSG are non-legally binding arrangements. Therefore, it could be even more necessary to comply strictly with the agreements. Consequently, legal improvements are required for effective implementations of Nuclear Cooperation Agreements. While it may be challenging to institutionalize details of 29 Nuclear Cooperation Agreements, it is essential to legally specify key elements such as the list of items subject to agreements, responsible authority, requirements and procedures for implement the agreement obligations, and penalties. Furthermore, domestic awareness on compliance with Nuclear Cooperation Agreements is lower compared to the system of export license for Trigger List Items. The continuous outreach is also necessary, along with institutional improvements.

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Keywords: Trigger list items, Export controls, Bilateral nuclear cooperation agreements, Foreign trade act, Nuclear safety act

Development of Nuclear Material Accountancy (NMA) Related Factors for Safeguardability Evaluation Method (SEM)

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The development of advanced nuclear facilities is progressing rapidly around the world. Newly designed facilities have differences in structure and operation from existing nuclear facilities, so Safeguards by Design (SBD), which applies safeguards at the design stage, is important. To this end, designers should consider the safeguardability of nuclear facilities when designing the system. Safeguardability represents a measure of the ease of safeguards, and representative evaluation methodologies are Facility Safeguardability Analysis (FSA) and Safeguardability Check-List (SCL). Those two have limitations in the quantification of safeguardability. Accordingly, in this study, the Safeguardability Evaluation Method (SEM), which has clear evaluation criteria based on engineering formulas, was developed.

Nuclear Material Accountancy (NMA), a key element of Safeguards, requires the Material Balance Area (MBA) of the target facility and performs Material Balance Evaluation (MBE) based on the quantitative evaluation of nuclear materials entering or leaving the MBA. In this study, about 10 factors related to NMA were developed, including MBA, Key Measurement Point (KMP), Uncertainty of a detector, Radiation signatures, and MUF (Material Unaccounted For). For example, one of the factors, MUF is used in MBA to determine diversion through analysis of unquantified nuclear materials and refers to the difference between Book Inventory and Physical Inventory, as well as errors occurring during the process in bulk facilities, errors in measurement, or intentional use of nuclear materials. This occurs in situations such as attempted diversion, and accurate MUF evaluation is essential for solid Safeguards implementation. MUF can be evaluated using the following formula ($MUF = (PB + X - Y) - PE$). The IAEA's Safeguards achievement conditions ($MUF < SQ$) should be met. Considering this, MUF-related factors were developed as follows. ($SEM = 1 - \frac{MUF}{SQ}$) In this way, about 10 factors were developed and described in the text.

This factors is expected to serve as an important factor in evaluating the safeguardability of NMA, and in the future, safeguardability factors related to Containment & Surveillance (C&S) and Design Information Verification (DIV) will be additionally developed to conduct a comprehensive safeguardability evaluation of the target facility. This methodology can significantly enhance safeguardability during the design stage of nuclear facilities.

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Keywords: Safeguardability, Safeguardability evaluation method (SEM), Non-proliferation, Safeguards-by-design, Safeguards

A Foreign Case Study on Cost Analysis of Systems for Recovering Uranium From Seawater

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To ensure the long-term supply and sustainability of uranium fuel, exploring alternative resources is essential, particularly considering that terrestrial reserves of uranium are limited (about 4.6 million tons). Since the amount of uranium dissolved in seawater is approximately 1000 times that of terrestrial reserves (i.e., about 4.5 billion tons), uranium extraction from seawater (UES) can be an alternative resource. However, the ultra-low concentration of uranium in seawater (about 3.3 ppb) poses a significant challenge in achieving economic feasibility for UES.

This paper introduces case studies on the cost analysis of systems for recovering uranium from seawater, specifically focusing on braided fiber-based adsorbents developed by JAEA and ORNL. The cost analysis has been conducted based on using the deployment of these adsorbents on the bottom of the sea, which is a passive deployment method, thereby reducing the total costs of recovery. The analysis results can be used to identify R&D areas necessary for reducing cost components, making UES economically feasible.

Keywords: Uranium extraction from seawater (UES), Cost analysis, ORNL, JAEA

Usability of Diversion Path Analysis Tool to Incorporate SBD Concept Into a New Nuclear Facility

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Safeguards systems and measures are determined through diversion scenario analysis based on the facility design information submitted to the IAEA when a new nuclear facility is introduced. While the concept of safeguards-by-design (SBD), which considers the safeguards from the design phase for a facility operator to minimize unplanned changes or disruption to facility operations as well as for the IAEA to increase the efficiency and effectiveness in safeguards implementation, has been emphasized for more than a decade, there is no practical tool or guidance on how to apply it. In this study, we develop a diversion path analysis tool and introduce how to apply SBD using it.

A diversion path analysis tool was developed based on the elements that constitute diversion and the algorithm generated based on the initial information of facility and nuclear material flow. The results of utilizing the analysis tool depending on a different level of facility information and the safeguards set-ups were compared through examples. Taking a typical light water reactor as an example, the test analyzed the automatic generation of dedicated routes, configuration of safeguards measures, and diversion path analysis. Through this, the application and limitations of the analysis tool are discussed, and ideas for utilization according to the SBD concept and necessary regulatory guidance are proposed.

The results of this study are expected to be directly utilized to domestic nuclear control during the regulation process for a construction of new nuclear power systems, and furthermore, to enhance national credibility in the engagement with the IAEA for implementation of safeguards.

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Keywords: Safeguards, Event tree, Fault tree, Diversion path, Algorithm

A Study on the Uncertainty of Radiochronometry Based on Bateman Equation

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Nuclear Forensics is recognized as an essential component in the nuclear non-proliferation verification sector by the international community. It is being advanced under the leadership of the IAEA, the U.S., and the EU. Both the U.S.'s Lawrence Livermore National Laboratory (LLNL) and the international collaborative organization, the Nuclear Forensics International Technical Working Group (ITWG), have proposed to establish a relationship between the production timing and radiochronometry of nuclear materials or samples to utilize in the field of nuclear forensics. Radiochronometry of nuclear materials is calculated based on the Bateman equation, incorporating factors with uncertainties derived from tests, experiments, and analyses. The results from the nuclear activity radiochronometry also encompass uncertainties, affecting their reliability.

This study examined the mathematical uncertainty calculations related to the results of nuclear activity radiochronometry, focusing on calculation methods, contribution rates per factor, and sensitivities. Uncertainty factors for the Bateman equation-based radiochronometry were observed in the decay constants for each nuclide type and the uncertainty in the radioactive ratio of the tracer nuclide. The sensitivity for each factor revealed that the uncertainty in the radioactive ratio of the signature nuclide contributed more significantly than the uncertainty in decay constants for each nuclide type. Each factor displayed a distinct sensitivity curve relative to the radioactive ratio. As it approaches a radioactive equilibrium, the sensitivity tends to increase infinitely, indicating a corresponding trend of infinite increase in uncertainty. Because the time and curve shape to reach radioactive equilibrium vary depending on the signature nuclide, it's essential to choose an appropriate signature nuclide based on the anticipated period and analysis requirements for nuclear activity radiochronometry. However, radiochronometry using mathematical methods is limited to the relationship between parent and daughter nuclides, presenting the potential for underestimation of uncertainty factors like decay constants. Future research will need to focus on uncertainty calculation methods through computational simulations, especially using the Monte Carlo method, to overcome the limitations of mathematical approaches and potential underestimations.

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Keywords: Nuclear forensics, Nuclear material, Radiochronometry, Decay chain, Uncertainty

Surface Temperature Analysis of Yongbyon Nuclear Complex Using Thermal Infrared Satellite Imagery

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This study presents a method for analyzing the surface temperatures of specific facilities, such as the 5 MWe reactor within the Yongbyon Nuclear Complex, to explore its potential utility in monitoring suspected nuclear-related activities in North Korea using thermal infrared (TIR) satellite imagery (Landsat series). TIR band data is utilized to derive surface temperatures in the specified areas, and the temperatures are analyzed on a monthly basis to examine any patterns within these regions. This research provides a pattern-of-life on temperature variation for the target areas through multiple TIR image datasets, offering additional information to analyze facilities' operational status in remote and inaccessible regions.

Acknowledgements

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Keywords: Thermal infrared (TIR) satellite imagery, Surface temperature, Nuclear activities

Study on On-Site Inspection Measures Between the US and Russia: Implications From the INF Treaty and START

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Arms control treaties during the Cold War generally used national technical means (NTM) to verify treaty compliance. This was because signatory states refused to agree on on-site inspection (OSI) measures since it would require some level of intrusion. Efforts on nuclear arms control such as the Limited Test Ban Treaty (LTBT) or Strategic Arms Limitation Talks (SALT) initially included some form of OSI but could not continue due to refusal from signatory states. The Intermediate-Range Nuclear Force (INF) treaty concluded between the US and the Soviet Union in 1978 was significant since both states agreed on a highly intrusive verification measure. The Strategic Arms Reduction Treaty (START) and the new START also called for OSI measures similar to the INF. Alongside reducing a significant number of nuclear warheads and limiting specific types of nuclear warhead delivery vehicles, these treaties also provided basic models for conducting on-site inspection (OSI).

OSI measures primarily rely on the political agreement between signatory states. However, the structure, types of inspections, number of inspections allowed, and technology/equipment used in each of the regimes also differ according to the objectives of each treaty. The INF treaty and START are salient cases as basic models for current nuclear disarmament verification research. Thus, this paper will conduct a case study on the procedures and mechanisms required for nuclear arms control verification in terms of OSI. Using the implications drawn from the INF treaty and START, this paper offers considerations for a potential nuclear disarmament verification.

Keywords: On-site inspection, Treaty compliance, Nuclear disarmament, Arms control

Study on Multilateral Nuclear Disarmament Verification Procedures and Technologies

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Any type of nuclear arms control or disarmament agreement requires some form of verification measure. Existing nuclear arms control treaties drew upon previous agreements such as the INF treaty, START, and IAEA nuclear safeguards inspections. However, previous treaties focused on limiting specific types of nuclear weapons and their delivery vehicles or reducing the total number of nuclear weapons rather than eliminating the nuclear enterprise as a whole. A potential nuclear disarmament verification treaty or agreement will depend on the geopolitical environment of the time as well as the national policies and priorities of each signatory state. Although research on the gradual reduction and eventual elimination of nuclear weapons is still ongoing, several states have cooperated to conduct experiments, exercises, and simulations on the procedures and technologies required for nuclear disarmament verification. Three of these efforts are the LETTERPRESS simulation conducted by the Quadrilateral Nuclear Verification Partnership (QUAD), NuDiVe Exercise conducted by the International Partnership for Nuclear Disarmament Verification (IPNDV), and the Menzingen experiment organized by the UNIDIR in partnership with the Swiss Armed Forces, Spiez Laboratory, Princeton University's Program on Science and Global Security, and the Open Nuclear Network.

These contain aspects for the development of a potential nuclear disarmament verification. The LETTERPRESS exercise conducted in 2017 tested potential activities and equipment inspectors might utilize in a nuclear weapon facility. The IPNDV NuDiVe exercises conducted in 2021 and 2022 tested the activities and equipment required for the verified dismantlement of a warhead within a dismantlement facility. Finally, the Menzingen experiment conducted in 2023 tested the practical procedures for the verification of a nuclear weapon's absence at a storage site. This paper will analyze the three cases to offer considerations on the procedures and technologies future nuclear disarmament verification might include.

Keywords: QUAD, IPNDV, Verification concepts, Verification technologies

Analysis of Spent Fuel Data and the Impact of Burnup and Cooling Time in Nuclide Source Strength

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Spent nuclear fuel continues to be generated domestically and abroad, and various studies are actively being conducted for interim dry storage and disposal of spent nuclear fuel. The characteristics vary depending on the type of spent nuclear fuel and the initial specifications, and based on these characteristics, it is essential to estimate the burnup and enrichment of spent nuclear fuel as a non-destructive assay. In particular, it is important to estimate the characteristics of spent nuclear fuel with non-destructive tests because destructive tests cannot be performed on all encapsulated spent nuclear fuel in case of intrusion traces in safeguards. Data is made by measuring spent nuclear fuel directly to evaluate burnup of spent nuclear fuel, but computer simulation research is also important to understand its characteristics because past burnup history is not accurately written, and destructive testing is difficult. In Sweden, the dependency of the burnup history in source strength and mass of light-water reactor-type spent nuclear fuel was evaluated, and this part was also applied to MAGNOX in consideration of the possibility of being used to verify DPRK's denuclearization.

SCALE 6.2 TRITON modeling was performed based on public information on DPRK's 5 MWe Yongbyon reactor, and the source strength of Nb-95, Zr-95, Ru-106, Cs-134, Cs-137, Ce-141, Ce-144, Eu-154 nuclides were evaluated. Since the burnup of MAGNOX is lower than that of light-water reactors, major nuclides in decay heat were not considered. The cooling period was evaluated based on 0, 5, 10, and 20 years. In case the discharge timing was different, the total period of discharge and reloading was the same, and the end-cycle burnup was the same, calculations showed that the source strength emitted from major nuclides was evaluated within 2-3% except for Ru-106 and Ce-144 nuclides. Even the burnup step of nuclear fuel is the same, and the reloaded length after discharge is different, i.e., the cooling period between is different at 5, 10, and 20, the source strength of Nb-95, Zr-95, Ce-144, and Cs-137 was evaluated as an error of 1%. Except for Ru-106 and Ce-144, nuclides are highly dependent on burnup. Compared to the case of light-water reactors, the possibility of a decrease in error needs to be considered later because the specific power is low.

As a result, radionuclides in released fuel depend on the effects of burnup, discharged and reloaded period, and a cooling period after release, and research is needed to correct the cooling period within the future burnup history. In addition, in this study, it is necessary to select a scenario-based burnup because the standard burnup due to the statistical treatment of discharged fuels was not considered as conducted in previous studies.

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Keywords: Source strength, MAGNOX, PWR, Burnup, Cooling

Analysis of Questionnaire Data for the Improvement of Physical Protection Education Curriculum and Training Program in KINAC

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Physical protection education was legislated by the Ministry of Education, Science and Technology (MEST) in November 2010. KINAC (Korea Institute of Nuclear Nonproliferation and Control) was designated as the exclusive institution for physical protection education and training by MEST in October 2011, and it has since functioned as the sole institution responsible for this critical aspect of nuclear security education in the country. Over the past decade, KINAC has undertaken a variety of training initiatives aimed at enhancing the capabilities of nuclear operators' physical protection personnel. Furthermore, it has consistently pursued annual curriculum revisions based on insights gleaned from surveys and workshops.

In conventional curriculum assessments, general surveys often rely on Likert scale or short-answer questions as primary indicators, mainly due to their ease of data processing. Descriptive questions, while capable of capturing diverse opinions, have historically been relegated to a secondary role owing to the inherent challenges associated with data analysis. While physical protection education has made concerted efforts to solicit diverse opinions through descriptive questions, difficulties in organizing and leveraging this valuable data have resulted in it primarily serving as reference material.

This study introduces a novel approach by employing ChatGPT, a chatbot, to conduct a comprehensive analysis of the descriptive questions from the physical protection education survey administered in the first half of 2023. The primary objective is to formulate a robust plan for curriculum enhancement based on a wide spectrum of opinions.

Following the completion of physical protection training by 2,014 individuals in the first half of 2023, a survey was distributed, yielding an impressive response rate of 95.7% with 1,927 respondents. Chatbots were harnessed to extract major keywords and perform frequency analyses on approximately 360 responses to descriptive questions in the survey. The analysis revealed that certain keywords emerged with notable frequency, in the following order: "drone" (mentioned 51 times), "access management" (mentioned 28 times), "inspection and search" (mentioned 27 times), and "cybersecurity" (mentioned 20 times). Further analysis of these major keywords and related content revealed a consensus among trainees that there is a pressing need to incorporate topics addressing drone threats and responses, as well as strategies to fortify access management into the curriculum.

This study underscores the potential to harness standardized data analysis techniques to synthesize and integrate trainees' subjective opinions, thereby providing a solid foundation for the refinement of the curriculum.

Keywords: Physical protection education, Chatbot, ChatGPT, Survey

Development of Environmental Geometrical Data Transfer (EGDT) for Maintaining CoK of Sample Extraction in Unreported Area

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Recently, the status of North Korea's denuclearization has become an international issue, and there are also indications of potential nuclear proliferation among neighboring countries. So, the need for establishment of nuclear activity verification technology and strategy is growing. In terms of ensuring verification completeness, sample collection-based analysis is essential.

The concepts of Chain of Custody (CoC) and Continuity of Knowledge (CoK) can be defined in the process of sample extraction as follows: CoC is interpreted as the 'system for managing the flow of information subjected by the examinee', and CoK is interpreted as the 'Continuity of information collection through CoC subjected by the inspector'. In the case of sample collection process in unreported areas for nuclear activity verification, there are additional risks such as worker exposure/kidnapping or sample theft/tampering. Therefore, the introduction of additional devices might be required to maintain CoC and CoK in the unreported area.

In this study, an Environmental Geometrical Data Transfer (EGDT) was developed to ensure the safety of workers and the CoC/CoK of the samples during the collection process. This device was designed for achieving both mobility and rechargeability. It is categorized into two modes based on its intended users: sample mode and worker mode. Through the sensors, which is positioned in the rear part of device, such as radiation, gyroscope, light, temperature, humidity and proximity sensors, it can be easily achievable various environmental information in real-time. Additionally, GPS information can also be received, allowing for responsiveness to various hazardous scenarios. Moreover, the OLED display positioned on the front gives us for checking device information such as the current status of the device such as the battery level, the connectivity of wifi, and etc. Finally, an alarm function was integrated to enable rapid awareness during emergency situations. These functions can be updated and modified through Arduino-based firmware, and both the device and the information collected through it can be remotely controlled via custom software.

Based on the presented design conditions, a prototype was developed and field assessments were conducted, yielding results within an acceptable margin of error for various scenarios. Through the application of the EGDT developed in this study to the sample collection process for nuclear activity verification purposes, it is expected to achieve a stable maintenance of CoC/CoK through more accurate information transmission and reception.

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Keywords: Nuclear activity verification, Sample extraction, Unreported area, Remotely-controlled, Real-time transmission and reception

Initial Schedule Management Direction for SMR Project

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In the case of nuclear projects, when developing a new reactor type, it is necessary to confirm the reactor type, secure the safety, and especially obtain the construction permit approval of the licensing authority for construction. Schedule management is necessary to carry out nuclear projects, and progress rate management of project progress management is largely composed of three elements: scope management, cost management, and resource management. However, in the case of the small modular reactor (SMR) project currently being carried out, it is difficult to calculate the progress rate including budget and resources due to the nature of the project. Therefore, in the SMR project, it took two years from the beginning to prepare the integrated project master schedule (IPMS) to prepare the draft, and then two revisions were made over a year and a half.

In this SMR project, we will consider the entire construction period such as design, purchase and production, construction, commissioning, and operation in terms of scope management. The entire document list was created using the document review and approval sheet created at the beginning of the design. In the PMIS (Project Management Information System), the number of approved documents was calculated by comparing the list of engineering documents. In the purchase production part, the main core equipment such as the primary system nuclear steam supply system (NSSS), the secondary system turbine and condenser, and the man machine interface system (MMIS) are managed. Purchasing and manufacturing management shall be managed so that major equipment can be delivered in a timely manner in accordance with the schedule for delivery of equipment in the IPMS. In order to prevent delays in the start of production, it is necessary to minimize the waiting time for work through advance management tasks such as insurance of drawing, stocking of materials, availability of production facilities, etc.

In this way, we decided to carry out the schedule management for the design, purchase and manufacturing part in the SMR project first, and the installation, construction and commissioning part will be prepared for the future schedule management.

Keywords: Integrated project master schedule (IPMS), Construction permit (CP), Project management information system (PMIS), Progress rate

Global Trend of Anomaly Detection From Satellites for Nuclear Verification –An Analysis of Publications

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As remote sensing measures, satellite imagery has played an essential role in verifying nuclear activities for decades. Starting with the first artificial satellite, Sputnik 1, in 1957, thousands of satellites are currently missioning in space. Since the 2000s, the level of detail in pixels of an image (spatial resolution) has been significantly improving, thereby identifying objects less than one meter, even tens of centimetres. The more things are identifiable, the wider regions become targets for observation. With the increasing number of satellites, computer vision technology is required to explore the applicability of algorithm-based automation.

This paper aims to investigate the R&D publications worldwide from the 1990s to the present, which have tried to apply algorithms to verify any clandestine nuclear activities or detect anomalies at the site. The versatile open-source publications, including the IAEA, ESARDA, US-DOE national laboratories, and universities, are extensively reviewed from the perspective of nuclear nonproliferation (or counter-proliferation). Thus, target objects for applications are essentially located in nuclear-related sites, and the source type of satellite sensors focuses on electro-optical images with high spatial resolution. The research trend over time by groups is discussed with limitations at the time in order to contemplate the role of algorithms in the field and to present recommendations on a way forward.

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Keywords: Satellite imagery, Remote sensing, Countering nuclear proliferation, Anomaly detection, Change detection, Computer vision

A Study on the Linkage With Accountancy Management to Improve Obligated Item Management

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Korea has an agreement for cooperation with 31 countries, including the United States, Canada, Australia, and Japan. Under the agreement, the obligated items must be used for peaceful purposes, comply with nuclear non-proliferation and international safeguards, and obtain prior consent of shipment in case of enrichment, reprocessing, retransfer. Among them, the United States, Canada, and Australia have signed Administrative Arrangements of Cooperation Agreements (Supplementary Arrangements in Canada) for the international transfer and annual reports of obligated items.

When operators submit an annual report, the government compiles and make the annual report based on the data. Ideally, the final report is submitted by the operator should be the national annual report, but in practice, discrepancies occur between sum of the operator's and government's. In order to resolve these problems and strengthen the linkage between exports control and safeguards, our institute has begun the project to develop an 'Obligation Tracking System for internationally controlled items (OTS)'.

It is believed that obligated items which are unnecessarily included or omitted in annual report could be managed properly by developing OTS for life cycle of the items such as import, disposal/termination or transfer to other countries. In case of nuclear material, especially, the characteristics of the facilities (e.g., bulk-handling facilities) must be considered and principles of fungibility, equivalence, and proportionality should be applied to materials.

In order to computerize these procedures, we would like to propose to adopt the format of Code 10 for obligated item management. Code 10 is the form of the annex to the Korea-IAEA safeguards agreement which includes all records of inventory changes, import/export, and domestic movement of nuclear materials.

It is expected to minimize discrepancies between operators' annual reporting data and national annual reporting and further contribute to enhancing national trust and nuclear transparency.

Keywords: Nuclear bilateral cooperation agreement, Obligated items subject to the agreement, Management system design, Database for obligated items, and export and import control, Safeguards

Future Nuclear Security Strategy and Policy Based on the Analysis of Nuclear Security Consultative Group Cases

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To mark the 70th anniversary of the alliance between South Korea and the United States, President Yoon Seok-youl of South Korea and President Joseph R. Biden of the United States convened at the White House, adopting the pivotal “Washington Declaration.” This significant act paved the way for the establishment and institutionalization of the ROK-US Nuclear Consultative Group (NCG). The NCG is envisioned as a mechanism to address North Korea’s nuclear threat, striving for nuclear sharing and a nuclear defense system, thereby alleviating concerns about nuclear security.

The NCG is perceived as a crucial advancement in the realm of ‘tailored extended deterrence’ on the Korean Peninsula. However, its operational scope and efficacy remain subjects of debate within South Korea. A comparative analysis with other consultative entities, such as NATO’s Nuclear Planning Group (NPG) and Extended Deterrence Strategy Consultative Group (EDSCG), raises questions about NCG’s unique contributions and potential functional overlaps.

Furthermore, the establishment of the NCG represents a notable progression in the strengthened ROK-US alliance. This progression coincides with the resumption of large-scale joint nuclear security military exercises under the new administrations of both nations. Anticipated future operations within the NCG framework encompass the continual deployment of strategic assets and the execution of nuclear simulation exercises. Such actions serve not merely as a deterrent message to North Korea but also aim to instill confidence in the US’s commitment to extended deterrence among the South Korean populace.

This study aims to highlight the significance and implications of the ROK-US Nuclear Consultative Group (NCG) through an exhaustive comparative analysis of existing nuclear security consultative bodies and pertinent nuclear security policies. Moreover, this research emphasizes strategies to boost the NCG’s effectiveness, the necessity for policy enhancements to foster South Korea’s nuclear security autonomy, and the importance of raising nuclear security awareness among the general public.

Keywords: Nuclear consultative group, Nuclear security strategy and policy, Extended deterrence

Development of a Detector Storage Chamber for Extreme Environment

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The use of nuclear materials for nuclear power generation is increasing worldwide, and the International Atomic Energy Agency (IAEA) has signed an agreement with countries using nuclear materials to prevent using military purpose through the Non-Proliferation Treaty (NPT) for the management of nuclear materials. Accordingly, all member countries manage nuclear material and equipment facilities under the treaty and are obligated to conduct safety measures such as inspection, containment, and surveillance in accordance with safety standards.

The equipment used in the inspection basically consists of a Scintillator type and a semiconductor detector type, and is mainly used for portable equipment to ensure the integrity of the equipment. In general, the operating environment of the detector guaranteed by the manufacturer is -10 degrees to 40 degrees due to poor resolution and electrical problems. However, in the case of an outdoor environment other than a laboratory environment, it is difficult to maintain the above temperature conditions. In particular, the internal temperature of the vehicle used for transport rises to more than 50 degrees in Korea, making the detector stored therein vulnerable.

In this study, a storage chamber for extreme environments was developed. The developed chamber compared the internal temperature by heating the external temperature. In addition, the performance before and after heating was compared by heating the radiation detectors HPGe, CZT, and NaI from -20 to 70 degrees Celsius while using the storage chamber. Our proposed chamber can play a key role in applications with good performance in complex environmental adaptability in their design.

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Keywords: Extreme environment, Retrofit, Storage chamber, Portable detector

Preliminary Investigation on Potential Challenges in Integrating Nuclear 3S (Safety, Security, and Safeguards) for Pyroprocessing

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Nuclear safety, security, and safeguards (nuclear 3S) are essential components for establishing robust nuclear environments. Nuclear safety is to protect public and environments from radioactive contamination, which can be caused in accidents. Nuclear security is to protect nuclear facilities from terrorism or sabotage, which related to physical attacks or insider threats. And nuclear safeguards is to protect nuclear materials from extortion by a state with a purpose of weaponizing activities. When a new nuclear facility is introduced, it is possible to save abundant amount of resources by considering nuclear 3S in an early stage (design phases).

Initially, the international atomic energy agency (IAEA) recommended safeguards-by-design (SBD) approach. The concept of SBD gradually expands to nuclear 3S-by-design (3SBD). Though there are differences in purpose and target subject, each nuclear ‘S’ is closely related with others. When introducing a certain technology or equipment in order to enhance one ‘S’, another ‘S’ also get affected. The effect can be synergies or conflicts. For instance, confidential information in nuclear security is required for a safeguards activity. On the contrary, inspection equipment for safeguards can be used for security.

Pyroprocessing is a technology for managing used nuclear fuels. As pyroprocessing is a back-end fuel cycle technology, a sensitive nuclear technology, safeguards has taken a large portion of nuclear 3S research in an effort to achieve international credibility and nuclear transparency. As mentioned, there are both synergies and conflicts in integrating nuclear 3S. In this study, we investigate potential challenges in applying nuclear 3S integration to pyroprocessing by addressing synergies and conflicts. This approach will suggest required supplementary methods to build the reliable pyroprocessing environment.

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Keywords: Nuclear 3S, Nuclear 3S integration, Pyroprocessing, Safeguards

Development of Contaminated Area Detection Equipment and CTBTO On-Site Inspection

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The CTBTO is the Comprehensive Test Ban Treaty Organization to ban all forms of nuclear testing (underwater, air, and underground) worldwide and was adopted at the UN's 50th annual general meeting in September 1996. As of September 2023, 187 out of 196 countries signed and 178 ratified. The Republic of Korea signed it in 1996 and ratified it in 1999. Several major Annex 2 countries still need to ratify it, and certain countries have not even signed it, so it has not come entry into force.

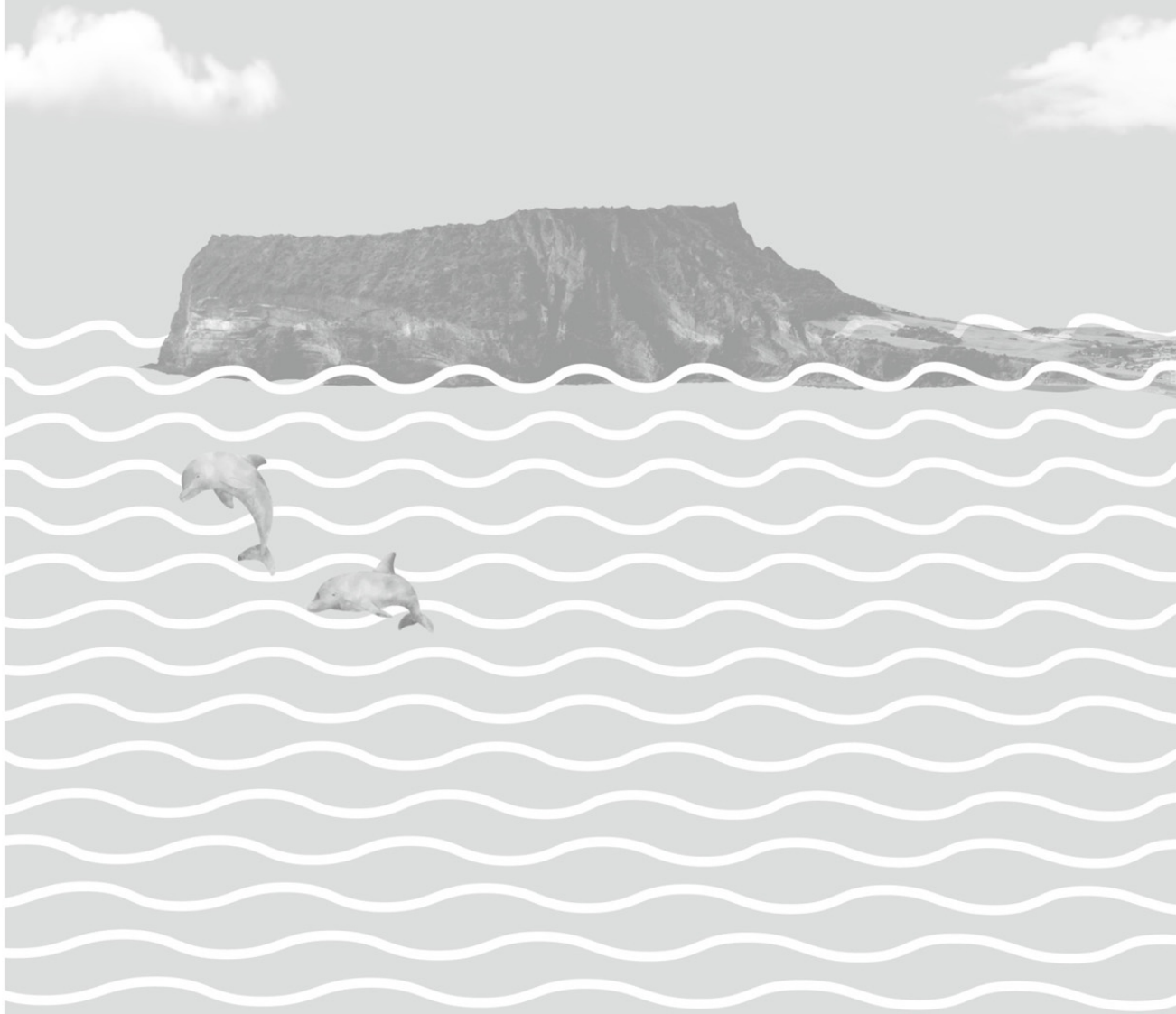
The CTBTO has three verification systems for nuclear tests and consists of the International Monitoring System (IMS), the International Data Center (IDC), and On-Site Inspections (OSI). IMS consists of seismic, hydroacoustic, infrasound, and radionuclide monitoring. The measured data are delivered to IDC, analyzed by CTBTO headquarters, distributed raw data, and analyzed forms to member states. The final means of verification is in the field of OSI and will be operated when CTBT takes effect. Based on the IMS data, inspectors will be dispatched to the Inspected State Party (ISP) to check for nuclear tests. KINAC is attending the Working Group B, OSI technology development verification along with KINS and KIGAM.

Since OSI is a means for final verification, integrated capabilities such as seismic and data interpretation and nuclides detection are required. CTBTO continues its efforts to foster integrated talent and modernize OSI equipment. Types of equipment include measurement, flight simulation equipment, and geographic information monitoring systems etcetera. KINAC is also developing equipment to detect contaminated areas using drones and probes. Development equipment is the nuclides detection and measurement of contaminated areas, and it is the equipment that prepares a control center and drops probes into suspected contamination areas to find a location of the radiation source. The probe can be used to track the location where the dose is most substantial through Bayesian estimation and source measurement.

Keywords: CTBTO, On-site inspection, Verification, OSI equipment

2분과

사용후핵연료 처분전관리 (ORAL)



Technical Aspects of Back-end Fuel Cycle of Advanced Accident Tolerant Fuel

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The Fukushima-Daiichi accident in 2011 revealed the limitations of Zr-alloys in accident scenarios where severe steam oxidation led to the liberation of heat and hydrogen and the destruction of the reactor core. In response to this accident, there has been a concerted effort by industry, national laboratories, and universities to develop cladding and fuel materials for light-water reactors (LWRs) that are more accident tolerant. The near-term approach has been to develop coatings for Zr-alloys that would provide additional safety and operational margin by virtue of its excellent corrosion/oxidation resistance at both normal and accident conditions. The designs being considered for implementation by major nuclear fuel suppliers include a thin Cr or a ceramic coating on the conventional LWR fuel cladding. For improved economics, the industries are also considering ATF coated cladding with high enrichment fuel (up to 8%) to achieve high burnup (> 75 GWd/MTU). While the development of ATF concepts (i.e., the front end of the fuel cycle), including coated claddings and doped fuels have progressed at an accelerated pace, relatively less attention has been devoted to the used fuel disposition of ATF fuels (i.e., the back-end of the fuel cycle). For accelerated deployment of the ATF designs in the current LWR fleet, it is necessary to investigate technical aspects of the ATF used nuclear fuel (UNF) management in transportation, storage, and disposal.

This presentation will provide a brief overview of state-of-the-art ATF developments and list out potential considerations to apply the fuels into back-end fuel cycle. New test plan should be planned to compare the characteristics of current LWR used nuclear fuels with those of the new fuel designs. For example, research focus can be understanding of ATF used fuel particulate size and quantity (at high burnup condition) and mechanical integrity of coated cladding under normal and off-normal conditions during transportation and long-term storage. Finally, the impacts of CRUD on the new fuel cladding, increased container weight, temperature, and radiation level to the back-end fuel cycle activities need to be investigated.

Keywords: Used nuclear fuel, Accident tolerant fuel, Transportation, Storage, and disposal

Rinsing Process of Residual Salt in a Metallic Fuel Produced Through Oxide Reduction for Its Enhanced Electrefining Suitability

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In pyroprocessing, the residual salts (LiCl containing Li and Li₂O) in the metallic fuel produced by the oxide reduction (OR) process are removed by salt distillation and fed into electrefining. This study undertook an investigation into the potential viability of employing a separate LiCl salt rinsing process as an innovative alternative to conventional salt distillation techniques. The primary objective of this novel approach was to mitigate the presence of Li and Li₂O within the residual OR salt of metallic fuel, subsequently facilitating its suitability for electrefining processes. The process of rinsing the metallic fuel involved immersing it in a LiCl salt environment at a temperature of 650°C. During this immersion process, the residual OR salt contained within the fuel underwent dissolution, thereby reducing the concentrations of Li₂O and Li generated during the OR process. Furthermore, the Li and Li₂O dissolved within the LiCl salt were effectively consumed through chemical reactions with ZrO₂ particles present within the salt. Importantly, even after the metallic fuel had been subjected to rinsing in a conventional LiCl salt solution, the concentration of Li and Li₂O within the salt remained consistent with its initial levels, due to the utilization of ZrO₂. Moreover, it was observed that the Li- Li₂O content within the metallic fuel was significantly diluted as a result of the rinsing process.

Keywords: Fuel cycle, Pyroprocessing, Electrolytic reduction, Electrefining, Rinsing process

Development of Model for Radionuclides Release Rate Calculation From a Transport Cask Submerged in the Deep Sea

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Given the situation in the Republic of Korea that all nuclear power plants are located at the seaside, the interim storage facility is also likely to be located at seaside and the maritime transportation of Spent Nuclear Fuel is considered inevitable. The Republic of Korea does not have an independently developed maritime transportation risk assessment code, and no research has been conducted to evaluate the release rate of radionuclides from a submerged transportation cask in the sea. Therefore, there is a need to develop a technology that can assess the impact of immersion accidents and establish a regulatory framework for maritime transportation accidents.

The release rate of radionuclides should be calculated from the flow rate through a flow path in the breached containment boundary. According to the cask design criteria, it is anticipated that even under severe accident conditions, the flow path size will be very small. Previous studies have evaluated fluid flow passing through micro-scale channel by integrating internal and external flows within and around a transport cask. As part of the evaluation, a comprehensive “Full-Field Model” incorporating external flow fields and a localized “Local-Field Model” with micro-scale flow paths were constructed. Sub-modeling techniques were employed to couple the flow field calculated by the two models. The aforementioned approach is utilized to conduct the evaluation of fluid flow passing through micro-scale flow paths.

This study aims to evaluate fluid flow passing through micro-scale flow paths using the aforementioned CFD (Computational Fluid Dynamics) method and aims to code the findings. The Gaussian Process Regression technique, a machine learning model, is utilized for developing a mathematical metamodel. The selected input parameters for coding are organized and their respective impacts are analyzed. The range of these selected parameters is tailored to suit domestic environments, and computational experiments are planned through Design of Experiments. The flow path size is included as an input parameter in the coded model. In cases where the flow path size becomes extremely small, making it impractical to use CFD techniques for calculations, Poiseuille’s law is employed to calculate the release rate. In this study, a model is developed to evaluate the release rate of radionuclides using CFD and mathematical equations covering the whole possible range of flow path size in a lost cask in the deep sea. The model will be used in the development of a maritime transportation risk assessment code suitable for the situation and environment in Korea.

Keywords: Spent nuclear fuel, Barrier effect model, Maritime transportation, Risk assessment, Release rate

A Study on the Thickness of the Oxide by the Characteristics of Spent Nuclear Fuel Burn-up

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The types of fuel loaded and burned in domestic nuclear power plants are WH-type and OPR/APR-type nuclear power plants, with a total of 19 types. In the case of spent nuclear fuel released in Korea, the low combustion level of 45,000 MWD/MTU or less accounts for about 75%. In terms of fuel type, WH 17×17 and CE 16×16 fuels account for about 85% of all spent nuclear fuels. The thickness of the oxide film of the fuel cladding can make the fuel rod vulnerable during reactor operation, directly affecting the integrity of the fuel rods. so, it is a very important design factor in design. Therefore, the fuel rod design code that predicts and evaluates this has also been developed to accurately predict fuel rod corrosion. And it's being applied to the design. In this study, the ECT probe measured by inserting it between fuel rods. The thickness of the fuel cladding oxide film was measured for spent nuclear fuel. When reloading operational nuclear fuel, the IAEA recommends an oxide film thickness of up to 100 micrometers. In this study, it was confirmed that spent nuclear fuels keeping integrity burned for 2-3 cycles were sufficiently maintained within the limit. However, the difference could be confirmed according to the characteristics of the coating material, the combustion cycle, and the use of poison rods. For the reliability of the data, symmetrical to the quadrant fuels were selected, and the fuel burned at the same period was measured. The method of selecting the target fuel can produce meaningful results.

Keywords: Spent nuclear fuel, Burnup, Oxidation, ECT (eddy current testing), Fuel rod, Fuel cycle

Revisit of Spent Fuel Transport Risk in Radiological Environmental Assessment for Major Nuclear Facilities in Korea

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In Korea, Kori Unit 1 and Wolsong Unit 1, have been permanently shut down in 2017 and 2019, and more nuclear power plants are expected to be permanently shut down after continued operation successively. Spent fuel has been generated during operation and stored in spent fuel pools. Due to the expected saturation of spent fuel pools within the next several decades, transportation of a huge amount of spent fuel is anticipated to interim storage facilities or final disposal facilities, even though the specific location is not decided.

The U.S. Nuclear Regulatory Commission (NRC) states that every environmental report prepared for the licensing stage of a Pressurized Water Reactor shall contain a statement concerning risk during the transportation of fuel and radioactive wastes to and from the reactor. Thus, the licensee should ensure that the radiological effects in accidents, as well as normal conditions in transport, do not exceed certain criteria or be small if cannot be numerically quantified. Specific conditions that a full description and detailed analysis of the environmental effects of transportation of fuel and wastes to and from the reactor are exempted are specified in 10 CFR Part 51.

Since there are no official requirements for radiological dose assessment for workers and public during the transportation of spent fuel in Korea, the margin when applying the U.S. regulatory criteria to the environmental impact assessment during the transport of spent fuel generated from domestic nuclear power plants is evaluated. A different approach would be needed due to the difference in the characteristics of spent fuel and geographical features.

Acknowledgements

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Keywords: Spent fuel, Transportation, Environmental effect, Margin

The Spent Fuel Information System Development Based on Semantic Technology

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In the establishment of procedures for managing spent fuel, the development of an information system for data management is an indispensable prerequisite. Given the prolonged period of spent nuclear fuel management, marked by numerous personnel changes and the anticipation of vast data retention, addressing this matter appropriately is imperative, particularly in the specialized field of spent nuclear fuel operations. Recognizing the need for a method to mitigate these challenges, we endeavored to apply semantic technology to the information system. To achieve this, we constructed the ontology of spent nuclear fuel and conducted research to transform it into a relational database. As a result, the information system, developed by the application of semantic technology, has attained the capability to comprehend and perceive relationships among information itself. Through this research, the system not only addresses previously identified concerns but also enhances its versatility, enabling it to perform functions previously unattainable within existing information systems.

Acknowledgements

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Keywords: Spent fuel, Relational database, Semantic technology, Ontology, RDF

A Preliminary Analysis of Safety Significance of Representative SSCs in Dry Storage Systems of Spent Nuclear Fuel

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In nuclear facilities, a graded approach is applied to achieve safety effectively and efficiently. It means that the structures, systems, and components (SSCs) that are important to safety should be assured to be high quality. Accordingly, SSCs that consist of nuclear facilities should be classified with respect to their safety importance as several classes, so that the requirements of quality assurance relevant to the designing, manufacturing, testing, maintenance, etc. can be applied.

Guidance for the safety classification of SSCs consisting of nuclear power plants and radioactive waste management facilities was developed by U.S.NRC and IAEA. Especially, in guidance for nuclear power plants, safety significance can be evaluated as following details. The single SSC that mitigates or/and prevents the radiological consequence or hazard was assumed to be failure or malfunction as the initiating event/accident occurred and the following radiological consequence was evaluated. Considering both the consequence and frequency of the occurrence of the initiating event/accident, the safety significance of each SSC can be evaluated. Based on the evaluated safety significance, a safety class can be assigned.

The guidance for the safety classification of the spent nuclear fuel dry storage systems (DSS) was also developed in the United States (NUREG/CR-6407) and the U.S.NRC acknowledges the application of it to the safety classification of DSS in the United States. Also, worldwide including the KOREA, that guidance has been applied to several DSSs. However, the guidance does not include the methodology for classifying the safety or the evaluated safety significance of each SSC, and the classification criteria are not based on quantitative safety significance but are expressed somewhat qualitatively. Vendors of DSS may have difficulties to apply this guidance appropriately due to the different design characteristics of DSSs.

Therefore, the purpose of this study is to evaluate the safety significance of representative SSCs in DSS. A framework was established to evaluate the safety significance of SSCs performing safety functions related to radiation shielding and confinement of radioactive materials. Furthermore, the framework was applied to the test case.

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Keywords: Spent nuclear fuel, Dry storage system, Graded approach, Safety classification, Safety significance

Investigation of Microstructural Characteristics for Circumferential and Radial Hydrides in Reactor-Grade Zirconium Cladding

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Hydride reorientation is widely known as one of the major degradation mechanisms in Zirconium cladding during dry storage. Some previous theoretical models for hydride reorientation used assumption of an ideal radial basal pole orientation for HCP structure of Zirconium cladding. Under this assumption, circumferential hydride was considered to precipitate in the basal plane while radial hydride was considered to precipitate in the prismatic plane, thereby giving energetical penalty on thermodynamical precipitation of radial hydrides. However, in reality, reactor-grade Zirconium cladding exhibits average 30° tilted texture, adding complexity to the hydride precipitation mechanism.

In this study, reactor-grade Zirconium cladding was charged with hydrogen and hydride reorientation-treated specimens were fabricated. Microstructural characterization of hydrides was conducted via following three methods in terms of interface and stored energy. And this study aimed to compare these characteristics between circumferential and radial hydrides. Using Electron Back Scattered Diffraction (EBSD), the interface was investigated assuming that interface lies parallel to the axial axis of the tube. These were further validated with Transmission Electron Microscope (TEM). In addition, Differential Scanning Calorimetry (DSC) analysis was conducted to calculate the stored energy.

This investigation is expected to establish fundamental understanding of how hydrides precipitate in Zirconium cladding with different orientations. And it will also increase the predictability of radial hydride formation and help understanding the mechanical behavior of Zirconium cladding with radial hydrides.

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Keywords: Hydride reorientation, Dry storage, EBSD, TEM, DSC, Microstructure

Methodology for Evaluation of Fracture Resistance of Irradiated Cladding Under Pinch Load

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The Spent Nuclear Fuel (SNF) cladding serves as the first barrier that prevents the release of radioactive materials. It is very important to maintain cladding integrity in SNF management. It is known that the pinch load applied to the cladding can lead to Mode-3 failure and the cladding becomes more vulnerable to this failure mode with the existence of radial hydrides and other forms of mechanical defects. In this study, a numerical analysis process was proposed to scientifically and systematically evaluate the fracture resistance of cladding with reoriented hydrides under pinch load. The mechanical behavior and fracture of the irradiated cladding under pinch load can be evaluated by Ring Compression Test (RCT). Under the stress field generated by RCT, the cracks propagate more easily through radial hydrides than circumferential hydrides. The δ -hydride which form within the α -zirconium matrix causes a large expansion strain due to the volume difference and voids form at the interface between the hydride and the zirconium matrix. Chan demonstrated that the load needed to form voids and separate the hard hydride precipitates from the Zr matrix is considerably lower than that which initiates brittle fracture of hydrides using a micro-cantilever test. Therefore, we propose a microstructure crack propagation analysis method based on Continuum Damage Mechanics (CDM) that can simulate fracture of hydride, zirconium matrix, and Zr/hydride interface. CDM is possible to simulate the hydride, zirconium matrix, and interface cracking in a continuum model based on cladding deformation. The RCT simulation model was constructed from the microscopic images of irradiated cladding. A pixel-based finite element model was created by separating the hydride, zirconium matrix, and interface using the image segmentation method on a morphology operation basis. The appropriate element size was selected for the efficiency of the analysis and crack propagation using CDM. The force-displacement curves and strain energy from RCT were compared and analyzed with the simulation results of different element sizes. The finalized RCT simulation model can be used to evaluate the fracture resistance of the irradiated cladding under the quantified pinch load and to establish the failure criterion of fuel rods under pinch load. The advantages and limitations of the proposed process are discussed.

Acknowledgements

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Keywords: Spent nuclear fuel, Irradiated cladding, Pinch load, SNF transportation, Crack propagation, Finite element analysis

The Effects of Hydrogen on Creep Behavior in Zircaloy-4 Cladding

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Zircaloy-4 is utilized in nuclear fuel rod cladding due to its strength and corrosion resistance. However, it can undergo deformation over time, known as creep, which poses a safety risk in reactors. Furthermore, hydrogen absorption during reactor operation can alter its properties and affect creep rates.

Previous research suggests a trend in which hydrogen concentration correlates unidirectionally with creep rates, either increasing or decreasing as the concentration rises. This trend can also be observed in EPRI's creep model, EDF-CEA Model-3. However, recent literature has suggested that creep behavior may vary depending on the state of hydrogen presence. Therefore, it has become evident that creep behavior can be influenced not only by hydrogen concentration but also by the state of hydrogen presence, 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 Zircaloy-4 plate 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. This investigation confirms that the state of hydrogen presence significantly impacts creep behavior within Zircaloy-4 cladding. As part of our additional research plans, we intend to conduct creep tests on the material based on its orientation, whether it is in the rolling direction (RD) or the transverse direction (TD). We also plan to perform creep tests on ring specimens. Additionally, for the ring specimens, we aim to evaluate how creep behavior differs between the cold-worked stress-relieved (CWSR) condition and the recrystallized annealed (RXA) condition achieved through high-temperature heat treatment.

Acknowledgements

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Keywords: Creep, Degradation, Zircaloy-4 cladding, Hydrogen concentration

Seismic Analysis of Free-Standing Spent Fuel Dry Storage Cask Considering Soil-Concrete Pad-Cask Interaction

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In the case of dry storage facilities, slipping of the cask or tip-over are dangerous phenomena. For this reason, in dry storage facilities, measures against slipping and tip-over or related safety evaluations are important.

Accidental conditions that can cause cask slippage and tip-over in dry storage facilities include natural phenomena such as floods, tornadoes, tsunamis, typhoons, earthquakes, and artificial phenomena such as airplane crashes. However, among natural phenomena, earthquakes are the most important natural phenomenon that causes tip-over. Also, many people had the stereotype that Korea is an earthquake-safe zone before 2016. However, earthquakes become a major disaster in Korea due to the 2016 Gyeongju earthquake and the 2017 Pohang earthquake, followed by the Goesan earthquake in October 2022.

In this paper, seismic analysis was performed based on dry storage facilities including multiple casks. Design variables for the construction of an analysis model for dry storage facilities were investigated, and seismic analysis was performed. To evaluate tip-over accident during earthquake, seismic load was used from 0.2 g PGA to 0.8 g PGA and these earthquakes were followed Design Response Spectrum (DRS) in RG 1.60. The friction coefficient of concrete pad was used from 0.2 to 1.0. As a result of the analysis, tip-over accident could not find in the analysis from 0.2 g to 0.6 g. However, tip-over was appeared at friction coefficients of 0.8 and 1.0 at 0.8 g PGA.

Tip-over angular velocity of cask was derived by seismic analysis and was compared with formula and tip-over analysis results. As a result, a generalized dry storage facility analysis model was proposed, and dry storage facility safety evaluation was conducted through seismic analysis. Also, tip-over angular velocity was derived using seismic analysis for tip-over analysis.

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Keywords: Concrete storage structure, Seismic evaluation, Earthquake, Tip-over, Concrete pad modeling

Scaled-Down Test Model Using CFD on the On-Site SNF Dry Storage Module

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Korea Hydro & Nuclear Power (KHNP) is currently developing a vertical concrete dry storage module for the dry storage of used nuclear fuel within nuclear power plants. This module is designed with a structure consisting of cylinders, which can block the ingress of external air, thereby preventing Chloride-Induced Stress Corrosion Cracking (CISCC). However, due to the presence of these cylinder structures, unlike conventional dry storage systems, it cannot directly dissipate heat to the external atmosphere, making thermal evaluation an important issue. The SF dry storage module being developed by KHNP is a massive concrete structure of approximately 20 m × 10 m × 7 m in size, employing a vertical storage system. To demonstrate the safety of such a large structure, there is no alternative to conducting experiments with scaled-down models. Furthermore, according to NUREG-2215 Section 5.5.4, it is explicitly mentioned that design-verification testing can be performed using scaled-down models. In this paper, a 1/4 scaled-down model was constructed to perform thermal performance verification experiments, and the effectiveness of this model was analyzed using Computational Fluid Dynamics (CFD) methods. The analysis results indicated that there was not a significant difference in terms of maximum concrete temperature and air outlet temperature. However, a considerable difference was observed in the canister surface temperature. Therefore, it is concluded that careful consideration of natural convection heat transfer is necessary for the full application of the scaled-down model.

Keywords: Safety validation tests, CFD, Scaled down model, Dry storage facilities

A Preliminary Study for Evaluation of Instant Release Fractions for PWR Spent Fuels Based on Literature Review

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While many countries consider direct disposal of the spent nuclear fuels, they need to consider long-term disposal scenarios with severe accidents such as the contact between underwater and the spent nuclear fuel due to large defect of the canister. Radionuclides releases rapidly with contacting water or slowly with dissolution of UO_2 matrix. The former is known as the ‘Instant Release’, and the latter is ‘Congruential Release’. Even though the instant release fractions (IRF) are much smaller than the congruential ones, IRF has to be treated carefully due to the fact that the instant releases lead to much larger value of the exposure dose rates than the congruential ones which proceed very slowly. It is known that the exposure dose rates by the instant releases are ~ 25 times larger than the one by the congruent release.

The radionuclides from UO_2 matrix migrate to the grain boundary, make bubbles, and make tunnels, which leads to instant releases of some radionuclides. The radionuclides in the gap between UO_2 pellet and cladding can be also instantly released. In addition, the radionuclides in the crud are instantly released. But in this paper, nuclides from the crud are not regarded, due to the lack of the leaching data. Meanwhile, there’re some nuclides that released from the construction materials like the cladding, the Rod Cluster Control Assembly (RCCA), or the other metal parts.

In this work, IRF values for major IRF nuclides such as Cs, I, Cl, Se for the reference PWR spent fuels of South Korea were evaluated based on the rationale from literatures’ review. In particular, these evaluations were done as the function of fission gas release (FGR), average discharge burnup, and fuel dimensions. In addition, the values of IRF for the other nuclides were also suggested based on the other institutes.

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Keywords: Instant release fractions (IRF), Fission gas release (FGR), Radionuclides, Grain boundary (GB), Gap, Spent nuclear fuel (SNF)

Structural Integrity Evaluation for Initial Characteristic Measurement Equipment and CANDU Spent Fuel

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A lot of CANDU Spent Fuels (CSFs) have been stored in spent nuclear fuel pools and dry storage facilities. In accordance with the enhanced nuclear regulations, the initial characteristics of CSF should be inspected to ensure the integrity of CSF and the reliable operation of storage system before loading it into a cask for long-term dry storage. For the inspections, an initial characteristics measurement equipment was designed, which is used for Pool-Side Examination (PSE) in the spent fuel pool of the pressurized heavy water reactor nuclear power plant.

Measurements using the equipment consist of non-contact inspections and contact inspections. The non-contact inspections do not affect CSF integrity, whereas the integrity of CSF can be reduced during the contact inspections under abnormal operating conditions because the probe of equipment may apply specific loads to the CSF. Therefore, the structural integrity evaluations of equipment and CSF are performed using Finite Element (FE) analyses for four combinations based on two abnormal conditions and two probe positions. The used abnormal conditions are the pressing load condition and the scratching load condition, and two probe positions are the center and bottom of the fuel rod in the longitudinal direction, respectively. In this evaluation, the bottoms of the fuel rod or CSF are defined as the regions facing the bottom surface of equipment.

The analysis of the pressing load condition is performed by pressing the probe of the equipment in radial direction of the CSF fuel rod. That of the scratching load condition is carried out by applying a specific radial load to the CSF fuel rod using the probe and then applying the load to the surface of the fuel rod while moving axially along the surface. All combinations are analyzed considering geometric, boundary and material non-linearity under the dynamic load, which is dependent on the equipment operating velocity.

The stresses of CSF and equipment components were obtained from these analyses. The maximum stress of each component was generated at the combination on the scratching load condition for the bottom position among the four combinations. The obtained maximum stresses are lower than the yield stress for each component material. Also, the CSF is not overturned due to the support plate of the equipment in all analyses. Therefore, the structural integrity and safety of the equipment and the CSF are maintained under abnormal operating conditions during the inspection using the initial characteristic measurement equipment.

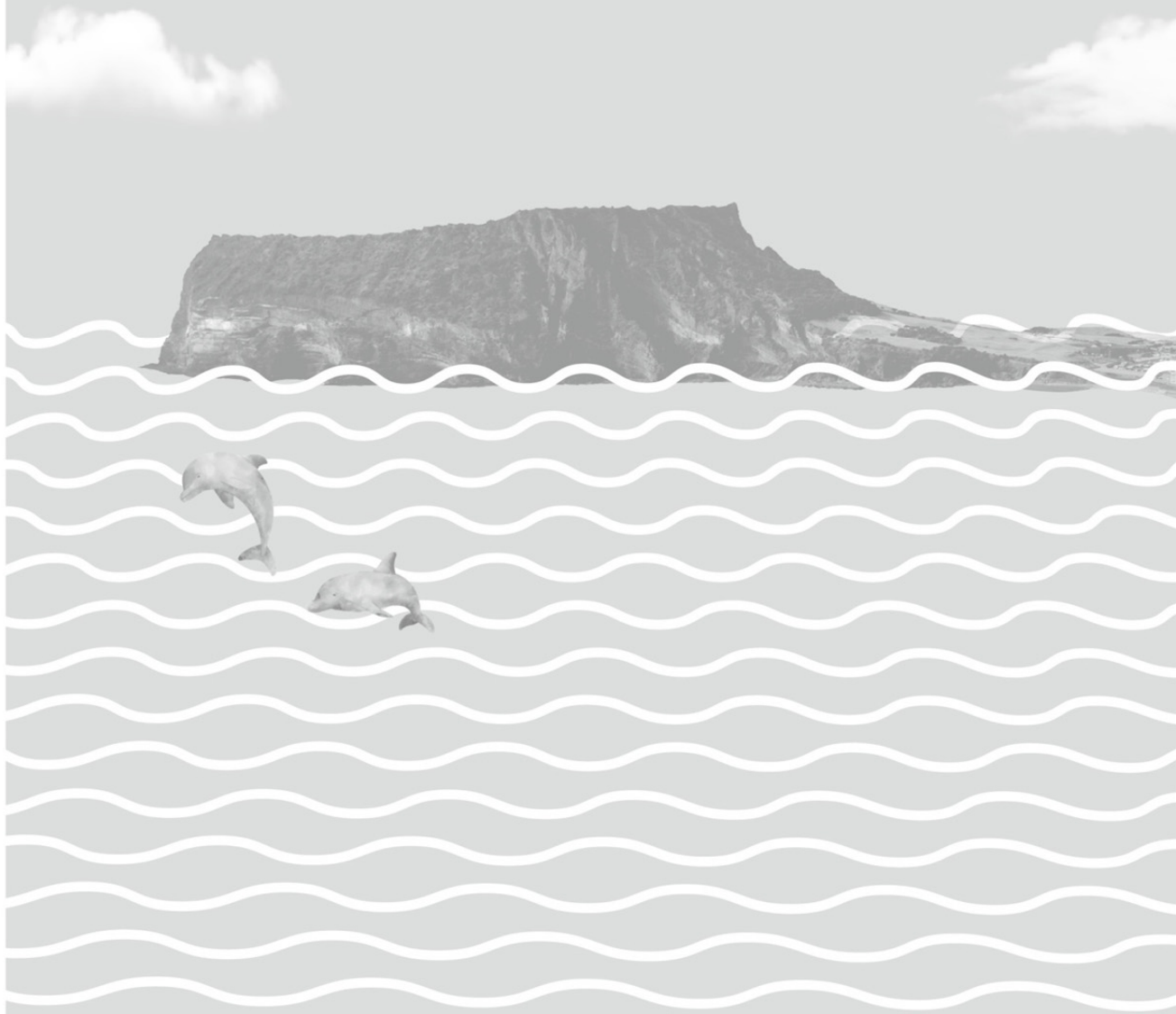
Acknowledgements

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Keywords: CANDU spent fuel (CSF), Initial characteristic measurement equipment, Abnormal operating condition, Non-linear dynamic FE analysis

2분과

사용후핵연료 처분전관리 (POSTER)



Simulation of Chlorination Process for Spent Nuclear Fuel Partitioning

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This study investigated the effectiveness of various chlorinating agents in partitioning light water reactor spent fuel, with the aim of optimizing the chlorination process. Through thermodynamic equilibrium calculations, the effects of using MgCl_2 , NH_4Cl , and Cl_2 as a single chlorinating agent or applying MgCl_2 , NH_4Cl , and Cl_2 sequentially for spent fuel chlorination were evaluated.

Furthermore, in this study, assuming the actual process operation situation, where only a part of the semi-volatile nuclides is removed during the heat treatment process, and including the process of precipitating the molten salt from the chlorination process with K_3PO_4 and K_2CO_3 precipitants, the percentage distribution of 50 nuclides in the light water reactor spent fuel into each process stream was quantitatively calculated using the simulation function of the HSC program and tabulated for intuitive viewing.

Compared to a single chlorinator, sequential chlorination more effectively separated the heat and radioactivity of the spent fuel from the uranium-dominated product solids. Specifically, the sequential application of the chlorinating agents following heat treatment led to a final solid separation characterized by 93.1% mass retention, 5.1% radioactivity, and 15.4% decay heat, relative to the original spent fuel. The findings underscore that sequential chlorination can be an effective method for spent fuel partitioning, either as a standalone approach or in combination with other partitioning processes such as pyroprocessing.

Acknowledgements

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Keywords: Sequential chlorination, Thermodynamic equilibrium, Spent nuclear fuel, Simulation, Partitioning

Heat Balance During Electrolytic Reducer Operation

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This study examined the heat balance in the electrolytic reducer during oxide reduction of pyroprocessing. The adoption of carbon anodes instead of conventional platinum anodes in the oxide reduction process has made it possible to apply high currents, and it has been observed that the temperature of the molten salt of in the reactor rises rapidly when applying high currents, so it is important to maintain an optimal operational temperature range.

In this study, salt resistant heat, reaction heat, and decay heat were identified as factors affecting heat balance during the operation of oxide reduction process. Equations describing the relationships among these factors were established. Then using this, a correlation was developed to understand the relationship between applied current and the molten salt temperature in the reactor observed in the actual operation of the carbon anode electrolytic reducer of KAERI.

Furthermore, this study proposed strategies to mitigate excessive temperature elevation during oxide reduction operation. A comparative assessment of these approaches was conducted. Considering KAERI electrolytic reducer operation environment, among the considered cooling strategies, the cooling effectiveness was calculated to be highest in the following order: heat transfer to extra salt, convection, conduction, argon gas bubbling.

Acknowledgements

This work was supported by National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2023).

Keywords: Electrolytic reducer, Heat balance, Heating, Cooling, Resistive heat

Surface Engineering Technologies for Mitigation and Repair of Chloride-Induced Stress Corrosion Cracking in Stainless-Steel Canisters for Spent Nuclear Fuel

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Safe management of spent nuclear fuel (SNF) is a key issue to determine sustainability of current light water reactor (LWR) fleet. However, none of the countries are actually conducting permanent disposal of SNFs yet. Instead, most countries are pursuing interim storage of spent nuclear fuels in dry cask storage system (DCSS). These dry casks are usually made of stainless-steels for resistibility against cracking and corrosion, which can be occurred over a long-term storage period. Nevertheless, some corrosion called Chloride-Induced Stress Corrosion Cracking (CISCC) can arise in certain conditions, exacerbating the lifetime of dry casks.

CISCC can occur if the three conditions are satisfied simultaneously: (i) residual tensile stress, (ii) material sensitization, and (iii) chloride-rich environment. A residual tensile stress is developed by the two processes. One is the bending process of stainless-steel plates into a cylindrical shape, and the other is the welding process, which can incur solidification-induced stress. These stresses provide a driving force of pit-to-crack transition. Around the fusion weld areas, chromium is precipitated at the grain boundary as a carbide form while it depletes chromium around it, leading to material susceptible to pitting corrosion. It is called sensitization. Finally, coastal regions, where nuclear power plants usually operate, tend to have a higher relative humidity and more chloride concentration compared to inland areas. This high humidity and chloride ion concentration initiate pitting corrosion on the surface of stainless-steels.

To prevent initiation of CISCC, at least one of the three conditions should be removed. For this, several surface engineering techniques are under investigation. One of the most promising approaches is surface peening method, which is the process that impacts the surface of materials with media (e.g., small pins, balls, laser pulse). By this impact, plastic deformation on the surface occurs with compressive stress that counteracts with pre-existing residual tensile stress, so this approach can prevent pit-to-crack transition of stainless-steels. Also, cold spray deposition can prevent CISCC. Cold spray deposition is a method of spraying fine metal powder to a substrate by accelerating them to supersonic velocity with propellant gas. As a result, a thin coating composed of the feedstock powders can protect the substrate from outer corrosive environments. In addition, the impact of the feedstock powder on the substrate during the process provides compressive stress, similar to the peening method.

Keywords: Spent nuclear fuel, CISCC, Surface peening, Cold spray deposition

Effects of Pore Characteristics on Tc Capture in Calcium Oxide Based Adsorbent

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Globally, the operation of nuclear power plants results in the production of a tremendous quantity of spent nuclear fuel. The methods for handling spent nuclear fuel can be categorized into three: storage, direct disposal and recycling. A technology designed to recycle accumulated spent nuclear fuel is pyroprocessing. In pyroprocessing, various fission products (FPs) such as C-14, H-3, I-129 and Cs-137 are generated. Among these FPs, technetium (Tc-99) is a gaseous nuclear isotope with a long half-life and high mobility in the form of TcO_4^- in aqueous solutions, making it essential to capture strictly in order to prevent radioactive contamination of the environment. In previous studies, ion-exchange or adsorption using MOFs (Metal Organic Frameworks) have been used to remove Tc-99. These methods, however, involve separation in aqueous solutions, not in the gaseous state.

In this study, we developed a CaO-based adsorbent for capturing Re as a surrogate for radioactive Tc-99. Isopropyl alcohol (IPA) was employed as a pore-forming agent during the preparation of the adsorbents, and its effects on characteristics and adsorption performance were investigated. The size of the pores were analyzed from nitrogen (N_2) adsorption isotherm analysis and mercury (Hg) intrusion curves. As a result, it was confirmed that the addition of IPA had a significant impact on the formation of macro-pores. Furthermore, this macroporous structure was found to enhance the adsorption performance of Re.

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

Carbonation of Strontium Chloride by Solid-Solid Reaction After Evaporation of LiCl-KCl Eutectic

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It has been investigated on the management of Strontium-90 in KAERI. It is needed to separate the solute from the salt solution for the recovery of strontium after the chlorination of the strontium oxide in molten salt. A vacuum distillation technology was used for the separation of strontium from the molten salt in our previous study. Strontium chloride was successfully carbonated by reactive distillation of $\text{SrCl}_2 - \text{K}_2\text{CO}_3 - \text{LiCl} - \text{KCl}$ system. In this study, it was tried to develop another route to recover strontium from the salt solution by a solid-solid reaction for avoiding the entrainment of product and the salt- K_2CO_3 reaction. Reactive distillation experiments were carried out for $\text{SrCl}_2 - \text{K}_2\text{CO}_3 - \text{LiCl} - \text{KCl}$ system. The carbonation temperature and pressure were 520°C and 0.8 bar. After the carbonation reaction, the temperature was elevated to 820°C to remove KCl from the reaction product. SrCO_3 and KCl peaks were found in the XRD analysis of the residual product. It could be concluded that SrCl_2 can be successfully carbonated after salt removal by the solid-solid reaction.

Acknowledgements

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Keywords: Strontium chloride, Carbonation, Recovery, Molten salt

Self Calibration of the Thermo-Mechanical Analyzer

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NFDC (Nuclear Fuel and materials Data Center) is designated as a one of the data center of National Standard Reference Center from Ministry of Trade Industry and Energy at Dec. 30 2008. The fields of designation were nuclear fuel and energy materials. NFDC produces standard reference data of nuclear fuel and materials.

To ensure reliability of experimental data uncertainty should be estimated. There are two kinds of uncertainty: A-type uncertainty from tester and B-type uncertainty from experimental equipments. To reduce the former, the measurement should be repeated for sufficient amount of times, and to reduce the latter type uncertainty all equipment have to be calibrated.

In this study self calibration process of thermo-mechanical analyzer (TMA) was established to ensure the B-type uncertainty. The self calibration was performed using the standard reference material and correction factor was obtained. The correction factor was defined as the ratio of the thermal expansion value of the standard reference material reported in the certificate and the thermal expansion value measured using TMA.

It is believed that the uncertainty evaluation process of TGA data developed in this study will be helpful for increasing reliability and stability evaluation of nuclear fuel and spent fuel.

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Keywords: Thermo mechanical analyzer, Standard reference data, Uncertainty, Self calibration, Nuclear fuel and materials data center

Two-Step Reactive Distillation Process to Mitigate Side Reaction Between the Salt and K_2CO_3

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Various disposal methods for spent nuclear fuels (SNFs) are being researched, and one of these methods involves separating high heat-generating nuclear isotopes such as Strontium-90 (^{90}Sr) and Cesium-137 (^{137}Cs) for deep disposal. These isotopes have relatively short half-lives and substantial decay energies. Especially, ^{90}Sr undergoes decay through Yttrium-90 to Zirconium-90, emitting intense heat with beta radiation. Therefore, the removal of these high heat-generating isotopes will significantly contribute to reducing disposal site area.

To remove ^{90}Sr from SNFs, molten salt was utilized in KAERI. During this process, it was discovered that ^{90}Sr dissolves in the molten salt in the form of SrCl_2 and/or Sr_4OCl_6 . Afterwards, it is crucial to recover ^{90}Sr in the form of oxide from the salt to create immobilized forms for disposal. This can be achieved by reactive distillation with K_2CO_3 . However, the amount of ^{90}Sr within the SNFs is only 0.121wt%, and even if all the ^{90}Sr in the SNFs were to leach into the molten salt, the quantity of ^{90}Sr in the molten salt would still be very small. Therefore, adding K_2CO_3 to the molten salt for reactive distillation could result in significant possibilities of side reactions occurring.

In this study, a two-step process was employed to mitigate the side reactions: the 1st step involves evaporating the all molten salts and the 2nd step includes adding K_2CO_3 to make oxides through solid-solid reaction. Eutectic LiCl-KCl , which is the most commonly used salt, was employed. The eutectic LiCl-KCl with SrCl_2 was heated at 850°C for 2 h to evaporate the salts under a vacuum (> 0.02 torr). However, after examining the distillation product before the solid-solid reaction, it was observed that SrCl_2 reacted with KCl in the salt, resulting in the formation of KSr_2Cl_5 . It means that salts containing KCl are not suitable candidates for reactive distillation aimed at producing immobilized forms. As an alternative, MgCl_2 could be a highly promising candidate because it is inert to SrCl_2 and according to a recent study in KAERI, MgCl_2 exhibited the most efficient separation of Sr among various salts. Therefore, we plan to proceed with the two-step reactive distillation using MgCl_2 for the future work.

Acknowledgements

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Keywords: Spent nuclear fuels disposal, Strontium-90, Reactive distillation, Recovery, Molten salt

Estimates of Spent CANDU Fuel Integrity During Dry Storage

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In KNF, fuel performance analysis modules were developed to predict the overall behavior of a fuel rod under normal operating conditions. Their main focus is to provide information on initial conditions prior to dry storage. Potential degradation mechanisms that may affect sheath integrity of spent CANDU fuel during dry storage are: creep rupture under internal gas pressure, sheath oxidation in air environment, stress corrosion cracking, delayed hydride cracking, and sheath splitting due to UO_2 oxidation for a defective fuel.

To upgrade the developed modules that address all the damage mechanisms, the first step was a review of the available technical information on phenomena relevant to fuel integrity. The second step was an examination of the technical bases of all modules, identify and extend the ranges of all modules to required operating ranges.

The 300°C spent CANDU fuel sheath temperature metric for dry storage ensures spent CANDU fuel element integrity from the failure mechanisms of creep rupture, oxidation and stress corrosion cracking at a failure probability of 2×10^{-5} for a dry storage time of 100 years. The 300°C sheath temperature metric for dry storage has relatively a lower failure rate than the target criteria for dry storage of spent LWR fuel. Although different modes of failure were treated separately for simplicity, ignoring possible synergistic effects, these results are conservative because of the conservative assumptions that have been made for evaluating spent fuel element conditions, and because of the inherent conservatism of the applied models. Additional conservatism of the model comes from the fact that isothermal conditions do not prevail in actual storage conditions.

Further R&D being considered includes acquisition of new functional models to implement overall fuel behavior evaluation and cover spent CANDU fuel in dry storage, and upgrades of the analysis module to achieve sufficient accuracy in key output parameters. The developed modules provide a platform for research and industrial applications, including the design of fuel behavior experiments and prediction of safe operating margins for spent CANDU fuel.

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Keywords: Spent CANDU fuel, Fuel performance analysis module, Potential degradation mechanism, Dry storage

Additive Effects on Corrosion in NaCl-MgCl₂ Molten Salt for Structural Materials

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Molten salt reactor (MSR) uses fluoride or chloride based molten salt as a coolant of the system, and fuel materials are dissolved in the molten salt, therefore it can be act as both coolant and nuclear fuel. A few issues have arisen from early-stage research and development program of MSR from Oak Ridge National Laboratory, including corrosion of structural materials and fission product management. For investigating the effect of additives on corrosion of structural materials, Mg(OH)₂ and MgCl₂*6H₂O are added into the NaCl-MgCl₂ eutectic salt. Prepared chloride salt is injected into the autoclave in the glove box, as well as corrosion coupons for candidate structural materials for molten chloride salt reactor, SS316, Alloy 600, and C-276 are also prepared. The temperature is set as 700°C. After 500 h corrosion experiment, the samples are taken out from the autoclave, and they are analyzed with scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDS). SS316 samples show weight loss with all salt conditions, while Alloy 600 and C-276 show weight gain after the corrosion experiment.

Keywords: Molten salt reactor, NaCl-MgCl₂, Corrosion, Mg(OH)₂, Moisture

Application of Roller Burnishing Treatment for Scaled Canister Model to Mitigate Chloride-Induced Stress Corrosion Cracking

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An austenitic stainless steel canister functions as a containment barrier for spent nuclear fuel and radioactive materials. The canister on the spent fuel storage system near the coastal area has several welding lines in the wall and lid, which have high residual tensile stresses after welding procedure. Interaction between austenitic stainless steel and chloride environment from a sea forms a detrimental condition causing chloride induced stress corrosion cracking (CISCC) in the canister. The South Korea is concerned with the dry storage of high-level spent nuclear fuel and radioactive wastes to be built on the site of a nuclear power plant. The importance of aging management has recently emerged for mitigating CISCC of dry storage canisters. When a corrosive pit is created by a localized corrosion in a sea water atmosphere, it initiates and grows as CISCC crack. Surface stress improvement works by inducing plastic strain which results in elastic relaxation that generates residual compressive stress. Surface stress improvement methods such as roller burnishing process can effectively mitigate the potential for CISCC of the canister external surfaces. The generation of compressive stress layer can inhibit the transition to cracking initiation.

In this study, a flat roller burnishing process was applied as a prevention technology to CISCC of stainless steel canisters. Roller burnishing process parameters have been selected for 1:3 scale canister model having a diameter of 600 mm, a length of 1,000 mm and a thickness of 10 mm on the basis of the burnishing conditions available to control residual tensile stress of austenitic stainless steel plate specimens. The surface roughness of the scaled canister model was investigated using a surface roughness measurement equipment after roller burnishing treatment. The surface residual stresses of the scaled canister model were measured by a hole drilling contour method attached with strain gauge. The burnishing test results showed that the surface roughness of the scaled canister model was considerably improved with flat rollers having the tip width of 4 mm. The surface of the scaled canister model had significant residual compressive stress after burnishing treatment. The roller burnished canister with good surface roughness could reduce the number of crack initiation sites and the residual compressive stress formed on the welded surface might prevent the crack initiation by reducing tensile residual stress in the weld zone, finally leads to CISCC resistance.

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Keywords: Spent fuel dry storage canister, Austenitic stainless steel, Chloride-induced stress corrosion cracking, Burnishing process, Scaled canister model

Observations for the SF Reduction Technics on the SNF Dry Storage Facility Using the Oxidation Treatment Method

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South It is necessary to develop the future technologies to improve the sustainability and acceptability of nuclear power plants generation. Currently, our company is preparing to build the dry storage facility on-site in accordance with the basic plan for managing high-level radioactive waste announced by the government in 2021. However, studies on technologies for the volume reduction of spent nuclear fuel to increase the efficiency of on-site spent fuel dry storage facilities are very not enough.

Accordingly, in this study, the storage efficiency and appropriateness for the SF volume reduction processing technologies such as SF oxide processing technology and consolidation technology are evaluated. Finally, the goal is to develop the optimized technologies to improve the storage efficiency of spent nuclear fuel.

As a result in this study is followings. [Safety] After removing volatile fission products (Xe, Kr, I, etc.), Xe, Kr, etc. are removed during storage of the sintered structures. UO_2 has a high melting point of approximately $1,000^\circ\text{C}$ after cesium (Cs) has been removed, and heat can be removed by natural convection. [Economy] 1999 DUPIC unit facility unit price reference, 2020 standard 328 \$/kg estimated. A Comprehensive Approach Considering the Whole System is needed. Benefit from replacement and continuous operation of metal storage containers. Changes in economic efficiency obtained in conjunction with fluctuations in electricity prices and disposal. [Waste filter] A separated solidification facility high-level waste filter is required, and overseas outsourcing must be considered. [Waste cladding]. Cannot be accommodated in low-level disposal site. This reason is why the Ni nuclides occur to be in bulk. [Metal structural material] It is possible to reduce the initial volume by 7.6% or more when compressed or melted, but the technology needs to be advanced. [Oxide blocks] Larger size and density are expected to improve storage and disposal efficiency. [Facilities operation waste] Expected to be able to be disposed of at mid-to-low level decommissioning sites in Gyeongju city. [Solidified volatile nuclides and activated metals] Expected to improve storage efficiency when used volume is reduced and stored, such as outsourced reprocessing. [Oxide block] Radioactivity and decay heat are estimated to be reduced by half during oxide treatment. 75% reduction in volume and 40% reduction in storage area compared to used nuclear fuel before treatment. [Merits/Shortages] Improvement of storage and disposal efficiency empirical research such as large-capacity [real-scale] oxide block production is required. Oxide processing facilities are likely to be classified as post-use nuclear fuel processing facilities. It is determined that additional documents such as a Radiation Environmental Report (RER) must be submitted. Existence of possible external leaks of glass, highly mobile radionuclides from the point of view of nuclear criticality and heat removal. Acceptancy requirements of citizens in the process of creating additional sites for oxide treatment facilities. Considering social public opinion, it is necessary to secure the acceptability such as residents' opinions convergence.

Characteristics of high nuclear non-propagation compared to other processing technologies involving chemical processing. Also, Expectation of volume reduction effect for spent nuclear fuel itself. Volume reduction methods for solid waste and gaseous waste are required.

Keywords: Spent nuclear fuel, Dry storage facility, SF reduction, SF oxidation treatment

Analysis of Crud on Thick Oxide Layer Formed on Zirconium Alloy Cladding

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It has been known that as oxide layer (ZrO_2) forms on the nuclear fuel cladding during irradiation in nuclear power plants, the corrosion kinetics are influenced by various parameters such as chemical environments.

One of those environments, crud deposition driven by coolant chemistry has an adverse effect on the formation of oxide (ZrO_2) and leads to increase thickness of the layer.

In this study, crud formation was performed through loop experiment equipment on the surface of intentionally-made oxide layer (ZrO_2) on cladding tubes and then the composition and characteristics of cruds were examined for the investigation of nuclear power plant environment.

As a result, various cruds in composition and microstructure were formed depending on the exquisite methods and conditions such as metal ion concentration.

Keywords: Oxide, Crud, Zirconium alloy

Analysis of Category of Accident Scenario for Spent Nuclear Fuel Dry Storage Facility

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In Korea, most temporary storage facilities for spent nuclear fuel are nearing saturation. As an alternative to this, the 2nd basic plan for high-level radioactive waste management specified the operation plan of dry interim storage facility. Meanwhile, the NSSC No. 2021-19 stipulates that it is necessary to evaluate the possibility and potential effect of accident before operating interim storage facility. Therefore, this study analyzed the categories of accident scenarios that may occur in dry storage facility as part of prior research on this. We investigated the case of categorization of dry storage facility accident scenarios of IAEA, NRC, KAREI, and KINS. The IAEA presented accident scenarios that could occur in on-site dry storage facility operated with silo and cask method. NRC has classified accident scenarios in dry storage facility and estimated the probability of accidents for each. KAERI and KINS selected major accident scenarios and analyzed the processes for each, in preparation for the introduction of dry storage facility in Korea in the future. Overall, a total of 10 accident scenarios were considered, and the scenarios considered by each institution were different. Among 10 scenarios, cask drop and aircraft collision were included in the categorization of most institutions. The results of this study can be used as basic data for cataloging accidents subject to safety evaluation when introducing dry interim storage facility in Korea in the future.

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Keywords: Spent nuclear fuel, Dry storage, Interim storage, Storage facility, Accident scenario

Enhanced Corrosion Resistance of Alumina Scale Grown on Haynes 214 Alloy for Aggressive Gas Corrosion Environments

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Pyroprocessing is a crucial method for recovering nuclear fuel materials, particularly uranium and transuranic elements (TRU), through electrochemical reactions in a LiCl or LiCl-KCl molten salt system, which is highly stable medium at elevated temperatures. In the electrochemical reduction stage, actinide metal oxides are effectively transformed into their metallic forms and retained at the cathode within a molten LiCl-Li₂O environment at 650°C. Simultaneously, oxygen ions (O²⁻) are generated at the cathode and then transported through the molten salt to be discharged at the anode, where they combine to form oxygen gas (O₂) on the anode's surface. One notable challenge in this electrochemical process is the generation of various byproducts during the anode oxide reduction step, including oxygen, chlorine, carbon dioxide, and carbon monoxide. Consequently, significant amounts of corrosion products tend to accumulate on the upper region of the anode's immersion area over time.

This report introduces a novel solution to mitigate corrosion-related challenges within the specified temperature range. We propose a selective oxidation treatment for the NiCrAl-based 214 Haynes alloy, involving exposure to 1,100°C in a reducing atmosphere. The objective is to stimulate the growth of protective α -Al₂O₃ scales on the alloy's surface. The resulting oxide scales have undergone thorough characterization using SEM, EDS, and XRD techniques. The pre-grown alumina scale has demonstrated commendable adherence and thermal stability, even when subjected to a chlorine-oxygen mixed atmosphere at the specified temperature.

Keywords: Pyroprocessing, LiCl-Li₂O, Electrochemical reduction, Haynes 214, Alumina scale

Evaluation of Changes in Ion Exchange Resin Performance According to Water Treatment Process Operation

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Korea Atomic Energy Research Institute's Post Irradiated Examination Facility safely stores spent nuclear fuel using a wet storage method to conduct research. Here, in order to remove the radioactivity released into the water, the stored water is passed through an ion exchange resin tower, and the radionuclides are exchanged with the bead-shaped ion exchange resin filled inside to lower the radioactivity concentration. At this time, because the stored water passes in one direction, clogging of the ion exchange resin occurs. If this phenomenon continues, the flow rate of the water treatment process decreases and operation efficiency decreases, so a backwashing process is necessary to re-mix the ion exchange resin and secure the flow rate again. In this study, the flow rate reduction trend according to the lifespan of the ion exchange resin and the flow rate recovery according to the backwash process operation amount were analyzed. The flow rate reduction trend of the ion exchange process was analyzed immediately after the backwashing process was started. In addition, the amount of flow recovery according to the backwash process operation amount was evaluated by the amount of waste generated during the backwash process and the number of days of operation until the backwash process was needed again. As a result, the flow rate of the ion exchange process decreased rapidly right after the backwash process until the position of the ion exchange resins was stabilized, and then stabilized. After that, it gradually decreased and reached the point where the backwash process was necessary. However, the decline trend was analyzed to be the same regardless of the lifespan of the ion exchange resin. In addition, the amount of waste generated during the operation of the backwash process was increased in the order of 400 L, 600 L, 1,100 L, 1,400 L, 3,500 L, and 4,200 L to increase the amount of operation of the backwash process. As a result, the number of days of ion exchange resin operation was 285 days, 338 days, and 342 days, was analyzed as 422 days, 322 days, and 720 days. Based on this study, it was confirmed that the flow rate reduction trend is the same regardless of the lifespan of the ion exchange resin, and as the backwash process operation increases, the number of days the ion exchange process can be operated increases, but there is a turning point where the waste treatment cost exceeds the number of days of operation.

Keywords: Water treatment process, Ion exchange, Ion exchange resin, Back wash, Radioactive liquid waste

Bending and Twisting Measurement of Spent Fuel

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More than 20,000 bundles of spent nuclear fuel are stored in the spent nuclear fuel storage pool of domestic nuclear power plants, and the dry storage facility project in the nuclear power plant site is being promoted as the saturation of the wet storage pool is imminent. Since bending or twisting of spent nuclear fuel is an important item in order to load spent nuclear fuel into a dry storage cask, PSE (Pool Side Examination) was performed to verify this. This paper describes whether it can be safely loaded into a dry storage cask based on the measurement results of bending or twisting of spent nuclear fuel.

The nuclear fuel assembly is designed to prevent excessive assembly bending and twisting because it can cause interference during dry storage and handling due to factors such as differences in depletion of nuclear fuel rods, irradiation growth, and coolant flow during reactor operation. The bending of the nuclear fuel assembly is measured by establishing a Plumb Line to photograph the nuclear fuel assembly based on it, and calculating a pixel that images the distance between the support grid and the Plumb Line. The twisting of the nuclear fuel assembly is measured by forming a virtual vertical plane with two Plumb Lines, and based on this, the twisting angle of the lower fixed compared to the upper fixed.

As a result of the measurement, the bending of spent nuclear fuel was about 0.0-10.2 mm, much lower than the reactor loading criteria of 15.0 mm, and in the case of twisting, about 0.0~2.2° much lower than the reactor loading criteria of 5.0°. Therefore, it was confirmed that spent nuclear fuel at domestic nuclear power plants was not affected by bending and twisting when loading into dry storage cask.

Keywords: Spent fuel, Dry storage, Burnup, Bending, Twisting

Agitated Dissolution Reaction for Nuclide Separation in Molten Salts

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The separation efficiency of nuclides in molten salt systems was investigated, with a focus on the influence of apparatus configuration and experimental conditions. A prior study revealed that achieving effective Sr separation from simulated oxide fuel required up to 96 hours, reaching a separation efficiency of approximately 90% using a static dissolution reaction in a porous alumina basket. In this study, we explored the impact of agitation on improving Sr separation efficiency and dissolution rates. The simulated oxide fuel composition consisted of 2wt% Sr, 3wt% Ba, 2wt% Ce, 3wt% Nd, 3wt% Zr, 2wt% Mo, and 89wt% U. To quantify the Sr concentration in the salt, we utilized ICP analysis after salt sampling via a dip-stick technique. Furthermore, we conducted ICP-OES analysis over a 55-hour duration to assess the separated nuclides. Complementing these analyses, SEM and XRD investigations were performed to validate the crystal structure and morphology of the oxide products.

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Keywords: Agitation effects, Molten salt, Nuclide separation, Simulated oxide fuel

Influence of Applied Potential and Salt Composition on Mixed Gas Composition Produced at Carbon Anode

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The potential use of cost-effective carbon anodes, as an alternative to expensive platinum, in the reduction of oxides within $\text{LiCl-Li}_2\text{O}$ molten salt at elevated cell potentials presents a promising avenue. However, this elevated potential gives rise to the generation of a complex mixture of anodic gases, including hazardous and corrosive species such as chlorine (Cl_2), oxygen (O_2), carbon monoxide (CO), and carbon dioxide (CO_2). In this study, we investigate the influence of applied potential and salt composition on the composition of the generated gas mixture. Real-time gas analysis was conducted during the TiO reduction reaction in the molten salt at 650°C using a MAX-300-LG gas analyzer. Simultaneously, electronic signals, including current, potential, and salt composition, were monitored throughout the oxide reduction process. Additionally, XRD investigations were performed to verify the crystal structure of the resulting products. This research provides valuable insights into optimizing carbon anode-based reduction processes for improved efficiency and safety.

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Keywords: Applied potential, Salt composition, Anodic gas analysis, Oxide reduction reaction, Carbon anode, Pyroprocessing

Verification of Burnup Reliability in Preparation for Dry Storage of SF

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Regulatory agencies require burn-up verification to ensure that dry storage casks using burn-up credit are not loaded with fuel with a reactivity greater than the allowable standard. Accordingly, in preparation for dry storage of SF, the reliability of the burnup was verified and action plans for fuel with confirmed errors were reviewed.

Reliability verification was performed by comparing the actual burnup calculated with combustion calculation code (TOTE, ISOTIN) used in NPP and the design burnup calculated with the nuclear design code (ANC). As a result of comparing the differences between actual burnup and design burnup for 7,414 assemblies of SF generated from CE-type NPPs, the average deviation was confirmed to be 0.79% and 220 MWD/MTU. In the CE-type NPPs, no fuel showing large deviations was identified, and it was confirmed that reliability was secured. As a result of comparing the differences in 11,082 assemblies of SF generated from WH-type NPPs, the differences were not large, averaging 1.16% or 422 MWD/MTU. However, fuels showing significant differences were identified, and cause analysis was performed for those fuels.

The cause analysis used a method of comparing the burnup of symmetrically loaded fuels in the reactor. For fuels that were not symmetrically loaded, a method was used to compare them with fuels with similar combustion histories. As a result of the review, it was confirmed that the fuel was under- or over-burned compared to symmetrically loaded fuel. For fuels for which clear errors have been identified, we are considering replacing them with the design burnup, and for fuels whose causes cannot be confirmed, we are considering ways to maintain the actual burnup.

Keywords: Burnup verification, Actual burnup, Design burnup, Dry storage

Phase Analysis of Uranium Oxides by XRD After Heat Treatment

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Once discharged, spent nuclear fuel undergoes an initial cooling process within deactivation pools situated at the reactor site. This cooling step is crucial for reducing the fuel's temperature. Once the heat has sufficiently diminished, two viable options emerge: reprocessing or interim storage. A method known as PUREX, for aqueous nuclear reprocessing, involves a chemical procedure aimed at separating uranium and plutonium from the spent nuclear fuel. This separation not only minimizes waste volume but also facilitates the reuse of the extracted materials as fuel for nuclear reactors. The transformation of uranium oxides through dissolution in nitric acid followed by drying results in uranium taking the form of $\text{UO}_2(\text{NO}_3)_2 + 6\text{H}_2\text{O}$, which can then be converted into various solid-state configurations through different heat treatments. This study specifically focuses on investigating the phase transitions of artificially synthesized $\text{UO}_2(\text{NO}_3)_2 + 6\text{H}_2\text{O}$ subjected to heat treatment at various temperatures (450, 500, 550, 600°C) using X-ray Diffraction (XRD) analysis. Heat treatments were also conducted on UO_2 to analyze its phase transformations. Additionally, the study utilized XRD analysis on an unidentified oxidized uranium oxide, UO_{2+x} , and employed lattice parameters and Bragg's law to ascertain the oxidation state of the unknown sample.

To synthesize $\text{UO}_2(\text{NO}_3)_2 + 6\text{H}_2\text{O}$, U_3O_8 powder is first dissolved in a 20% HNO_3 solution. The solid $\text{UO}_2(\text{NO}_3)_2 + 6\text{H}_2\text{O}$ is obtained after drying on a hotplate and is subsequently subjected to heat treatment at temperatures of 450, 500, 550, and 600°C. As the heat treatment temperature increases, the color of the samples transitions from orange to dark green, indicating the formation of different phases at different temperatures. XRD analysis confirms that uranyl nitrate, when heat-treated at 500 and 550°C, oxidizes to UO_3 , while the sample subjected to 600°C heat treatment transforms into U_3O_8 due to the higher temperature. All samples exhibit sharp crystal peaks in their XRD spectra, except for the one heat-treated at 450°C. In the second experiment, the XRD spectra of the heat-treated UO_2 consistently indicate the presence of U_3O_8 rather than UO_3 , regardless of the temperature. Under an oxidizing atmosphere within a temperature range of 300 to 700°C, UO_2 can be oxidized to form U_3O_8 . In the final experiment, the oxidation state of the unknown UO_{2+x} was determined using Bragg's law and lattice parameters, revealing that it was a material in which UO_2 had been oxidized, resulting in an oxidation state of $\text{UO}_{2.24}$.

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Keywords: Uranium dioxide, Oxidation, U_3O_8 , Back-end fuel cycle

Quantification Tests of Residual Moisture After Vacuum Drying

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Measuring the amount of water remaining in the canister after drying is critical to ensuring the integrity of Dry Storage. There are many ways to measure residual moisture, but dew point sensors are typically used to measure residual moisture after drying the canister. Because the dew point temperature inside the canister depends on the water vapor partial pressure, the water vapor partial pressure present in the canister can be determined using the dew point temperature. The British Standard (BS1336) proposes a formula for converting dew point temperature into vapor partial pressure. It is possible to validate changes in residual water concentration throughout drying and at the end of drying. It has around 500 ppmv when the dew point temperature hits -73°C at 3 torr.

Nuclear Regulatory Commission (US NRC) presented at 3 torr for 30 minutes as a criterion for the suitability of spent nuclear fuel drying. When the canister's internal pressure is around 1,000 torr and the dryness criteria are met, the moisture concentration for this value is around 3,000 ppmv. We conducted a vacuum drying test of a 57 liter test vessel. It is filled with helium after vacuum drying was completed, and the concentration of residual water is measured by AquaVolt Moisture Analyzer (AMA) connected by a sample flow line. After the vacuum pressure of 1.5 torr was reached, the test vessel was filled to a pressure of 1,140 torr of helium after 30 minutes. The average temperature inside the basket inside the test vessel is 50°C, the dew point temperature is below -70°C, the pressure of test vessel is around 1,000 torr, and the measurement results of the AMA connected to the sample line showed less than 200 ppmv. From these results, we can evaluate that the residual moisture in the test vessel is about 0.01 gram.

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

Reaction of ZrCl_4 and LiCl-KCl Molten Salt

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It is known that ZrCl_4 can be used in the chlorination process of spent nuclear fuel. However, its solubility in high temperature molten salt is very small, making it difficult to dissolve a large amount of ZrCl_4 . Therefore, in this study, a flange-type sealed reactor was manufactured to observe the reaction characteristics of LiCl-KCl salt and ZrCl_4 . LiCl-KCl salt and ZrCl_4 were placed in each alumina crucible, the reactor was sealed, and heated. The temperature at the reactor surface was above 500°C and maintained at that temperature for 48 hours. After completion of the reaction, the reactor was opened and the reaction products were recovered from each alumina crucible. The crystal structure of the reaction product was identified through XRD analysis, and the concentration of Zr was analyzed using ICP. Reaction characteristics were observed according to the molar ratio of ZrCl_4 added to the number of moles of KCl in LiCl-KCl salt. The molar ratios of ZrCl_4 to KCl were 0.5, 1, 2, and 3, respectively. As a result of each experiment, more than 95% of the injected ZrCl_4 was vaporized and there was almost no residue in the ZrCl_4 crucible. In the LiCl-KCl crucible, the weight increased in proportion to the amount of ZrCl_4 added. As a result of XRD analysis, K_2ZrCl_6 was confirmed in all LiCl-KCl salt product. When the ZrCl_4/KCl molar ratio was 2 and 3, LiCl-KCl could not be confirmed. Additionally, when the ZrCl_4/KCl molar ratio was 1, LiCl was identified, but KCl was not found. Almost all of the KCl appears to have reacted with ZrCl_4 . ICP analysis results showed that the Zr concentration was proportional to the amount of ZrCl_4 added in each LiCl-KCl salt, and exceeding the number of moles of reaction with KCl in the LiCl-KCl salt was observed. Therefore, these experimental results showed that ZrCl_4 can be dissolved in LiCl-KCl salt at a maximum concentration higher than its solubility.

Acknowledgements

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Keywords: Zirconium chloride, K_2ZrCl_6 , Molten salt, Dissolution, Chlorination

Intentional Aircraft Crashes on the On-Site SNF Dry Storage Facilities

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Since the September 11 terrorist attacks in the United States, concerns about intentional aircraft crashes into nationally critical facilities have soared in countries around the world. The United States government advised nuclear utilities to strengthen the security of nuclear power plants against aircraft crashes and stipulated aircraft crash assessment for new nuclear facilities. Interest in military missile attacks on nuclear facilities has grown after Russia attacked Ukraine's Zaporizhzhia nuclear power plant, where spent nuclear power dry storage facility is operated. Spent nuclear fuel dry storage facilities in nuclear power plant sites should also strengthen security in preparation for such aircraft crashes. Most, but not all, spent nuclear fuel dry storage facilities in Europe, Japan and Canada are operated within buildings, while the United States and Korea operate dry storage facilities outdoors. Since all of Korea's dry storage systems are concrete structures vulnerable to crash loads and are exposed to the outside, it is more necessary to prepare for aircraft crash terrorist attacks due to the Korea's military situation. Residents near nuclear power plants are also demanding assessment and protective measures against such aircraft crashes. However, nuclear power plants, including spent nuclear fuel dry storage facilities, are strong structures and have very high security, so they are unlikely to be selected as targets of terrorism, and spent nuclear fuel dry storage systems are so small that aircraft cannot hit them accurately. Collected opinions on the assessment of aircraft crash accidents at spent nuclear fuel dry storage facilities in nuclear power plant sites were reviewed. In addition, the current laws and regulatory requirements related to strengthening the security of new and existing nuclear power plants against intentional aircraft crashes are summarized. Such strengthening of security can not only ensure the safety of on-site spent nuclear fuel dry storage facilities, but also contribute to the continuous operation of nuclear power plants by increasing resident acceptance.

Keywords: Intentional aircraft crash, Terrorist attack, Missile attack, Security, Dry storage facilities

Suggesting Additive Manufacturing on Neutron Absorber Fabrication for Efficient Spent Nuclear Fuel Storage Capacity

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On a global scale, the storage of spent nuclear fuel (SNF) within nuclear power plants (NPP) has become an important research topic due to limited space caused by approaching capacity saturation. SNF have e been collected over decades of NPP operation, coming up to capacity limitation. In case of Korea, every reactor except Saeul 1 and 2 has reached a SNF storage saturation rate of over 75%. One of the most studied methods for enhancing storage capacity efficiency involves increasing storage density using racks with neutron absorbers. Neutron absorbers like borated stainless steel (BSS) are utilized to manage the reactivity of densely stored SNF. However, major challenges of applying BSS are manufacturing hardness from heterogenous microstructure and mechanical property degradation from helium bubble formation. This study suggests that innovative fabrication methods of 3D printing can be good candidate for easier fabrication and better structural integrity of BSS.

Directed energy deposition (DED), one of the 3D printing methods have become major candidate method for various alloys. It deposits alloy powder on base melt surface by high intensity laser, similar with welding process. Powder manufacturing is already demonstrated superior performance compared to casting in ASTM-A887, such as increased mechanical properties, owing to its well distributed chemistry of alloy. Moreover, as its original microstructural property, the formation of micro-pores through DED could lead to long-term performance improvements by capturing helium generated from the neutron absorption of boron. The potential for fabricating complex structure is also among the advantages of DED-produced neutron absorbers.

Expected challenge on DED application on BSS is lack of printing condition data, because the 3D printing process have to be kept very careful variables of thermal intensity, powder flux and etc. These processes may get through much of trial & error for initial condition approaching. Nonetheless, as a recommendation of improved neutron absorber for efficient SNF pool storage, the concept of 3D printed BSS stands out as an intriguing avenue for research.

Acknowledgements

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Keywords: Spent nuclear fuel, Fuel cycle management, Storage capacity, Neutron absorber, 3D printing

The Comparison of PHWR's Damaged Fuel Encapsulation System

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After the decision of the Wolsong unit 1 permanent shutdown (2019), spent fuel stored in the spent fuel bay (hereafter, SFB) should be transported to a dry storage facility (MACSTOR or Canister) in order to decommission Wolsong unit 1. Accordingly, KHNP has established a shipment schedule for damaged fuel of Wolsong Unit 1 and is trying to complete the shipment according to the schedule. Wolsong is equipped with transportation casks and dry storage facilities, but baskets need to be manufactured separately. In addition, license approval is required for baskets, transport cask, and dry storage facilities for legal grounds to contain, transport, and store damaged fuels. In this paper, the initial model, upgrade model, and automation model of encapsulation equipment planned to be introduced in Canada to handle PHWR's damaged fuel were compared, and the optimal model was selected in consideration of KHNP's planned spent fuel shipment schedule. The PHWR's damaged fuel encapsulation system is a system developed the PHWR's damaged spent fuel to be handled in the same way as the existing PHWR when storing it in the dry storage facility and loading a basket for capsulation into transport cask. At the Gentilly-2 nuclear power plant in Canada, a manually operated encapsulation system was used due to the low quantity of damaged fuel, which can be encapsulated two bundles a day, and this model is an initial model. In the case of Wolsong Unit 1, it has about 300 damaged fuels, so it takes about nine months to work when using the initial model. The upgrade model developed to improve work efficiency and reliability has increased work efficiency through some automation, but it would take about eight months to process the damaged spent fuels of Wolsong Unit 1, and this model has not yet been manufactured and applied. Lastly, the automation model changed the work location outside the SFB and automated drainage/drying operations. It is easy to maintain and replace consumables because the work is carried out by lifting the damaged fuel to a shuttle outside the SFB surrounded by a shielding chimney. Considering the reduction of drainage/drying time, it is possible to save about four times as much time as the initial model. That is, if the automation model is used, it is judged that the supply of Wolsong Unit 1 can be processed in about two months. However, in terms of license, initial model and upgrade model are expected to be easier and the period is expected to be shortened. However, if licensing is carried out as soon as equipment design is completed, it is believed that the period can be shortened by parallel equipment manufacturing and licensing. It is judged that the best way to comply with the target schedule is to select an automation model with excellent work performance, develop equipment, and proceed with licensing at the same time. Accordingly, KHNP is in the process of designing equipment with the aim of using the automation model to take out damaged fuel for Wolsong Unit 1.

Keywords: SFB, Damaged fuel, Spent fuel bay, Encapsulation, Dry storage facility

A Study on the Integrity and Behavior of Spent Nuclear Fuel in Temporary Storage

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Due to the saturation of spent fuel pool of nuclear power plant in Korea, temporary storage for spent fuel will be installed, and spent fuel will be stored and managed in dry cask for a considerable period of time. Since spent nuclear fuel must withstand continuous decay heat, radiation and high internal pressure of the fuel rod in the cask, behavior of spent nuclear fuel is needed to be reviewed.

Spent nuclear fuel used in Pressurized Water Reactor (PWR) in Korea is stored in a wet storage currently, but it is going to store a temporary dry-storage facility on Kori site. Therefore, it is very important and meaningful to evaluate the behavior of nuclear fuel with realistic modeling. Also, domestic PWR nuclear fuel has various burn-up. In the past, the burn-up of nuclear fuel in light water reactors was low, but in order to increase power generation efficiency, the concentration of uranium was increased and the number of new fuel was increased. Therefore, a large amount of nuclear fuel with burn-up of 45,000 MWD/MTU or higher, generally called high burn-up, is also stored in the spent fuel pool (SFP). Therefore, it is necessary to evaluate by dividing three different burn-up such as, low, medium, and high burn-up.

Thus, this study will review the behavior of nuclear fuel at different burn-up during the temporary storage period with FALCON (EPRI), computational code and analyze the factors affecting the integrity of nuclear fuel, including when the temporary storage is extended its additional lifetime. And this evaluation will contribute developing the spent fuel management plan in Korea.

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Keywords: Spent nuclear fuel (SNF), Spent nuclear fuel behavior, Dry cask, FALCON

Effect of Performance Change in Unit Process of Radionuclide Management Process on Waste Management Strategy

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The radionuclide management process is a conditioning technology to reduce the burden of spent fuel management, and refers to a process that can separate and recover radionuclides having similar properties from spent fuels. In particular, through the radionuclide management process, high heat- emitting, high mobility, and high toxicity radionuclides, which have a significant impact on the performance of disposal system, are separated and managed. The performance of disposal system is closely related to properties (decay heat and radioactivity) of radioactive wastes from the radionuclide management process, and the properties are directly linked to the radionuclide separation ratio that determines the composition of radionuclides in waste flow.

The Korea Atomic Energy Research Institute have derived process flow diagrams for six candidates for the radionuclide management process, weighing on feasibility among various process options that can be considered. In addition, the GoldSim model has been established to calculate the mass and properties of waste from each unit process of the radionuclides management process and to observe their time variations. In this study, the candidates for the radionuclide management process are evaluated based on the waste mass and properties by using the GoldSim model, and sensitivity analysis changing the separation ratio are performed. And the effect of changes in the separation ratio for highly sensitive radionuclides on waste management strategy is analyzed. In particular, the separation ratio for high heat-emitting radionuclides determines the period of long-term decay storage.

Keywords: Radionuclide management process, GoldSim modeling, Sensitivity analysis

Regulatory Analysis of Foreign Deep Geological Disposal Using Text Mining and Verification Standard Development

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EU taxonomy requires to solve problems for safe management of radioactive waste and disposal of spent fuel, which is a precondition for growing demand for nuclear power plant. Currently, Korea manages about 18,000 tons of high-level radioactive waste at temporary storage facilities in nuclear power plant sites, but such temporary storage facilities are expected to become saturated sequentially from 2031. Therefore, it is necessary to secure a permanent disposal facility to safely treat high-level radioactive waste. In accordance with the second basic plan for high-level radioactive waste management in 2021, it is necessary to establish requirements for regulatory compliance for the site selection and site acquisition, investigation and evaluation, and construction for the establishment of a deep geological disposal facility.

In this study, we analyzed the regulatory policies and cases of leading foreign countries related to deep geological disposal facilities for high-level radioactive waste disposal waste such as IAEA, USA, Sweden, and Finland using data analysis methodology. To analyze a large amount of text-based document data, text mining is applied as a major technology and a verification standard that secures validity and safety based on the regulatory laws described so far is developed to establish a regulatory base suitable for domestic deep geological disposal status.

Based on the collected data, preprocessing and analysis with Python were performed. Keywords and their frequency were extracted from the data through keyword analysis. Through the measured frequency values, the contents of the objects and elements to be regulated in the statutory items were grasped. And through the frequency values of words co-occurring among different sections through the analysis of related words, the association was obtained, and the overall interpretation of the data was performed.

The results of analyzing regulations of major foreign countries using text mining are visualized in charts and graphs. Word cloud can intuitively grasp the contents by extracting the main keywords of the contents of the regulations. Through the network connection graph, the relationship between related words can be visually structured to interpret data and identify the causal relationship between words. Based on the result data, it is possible to compare and analyze the factors to be supplemented by analyzing domestic nuclear safety case and regulations.

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Keywords: Deep geological disposal, High-level radioactive waste, Data analysis, Text mining, Safety regulation development

Enhancing Operational Stability in Pyroprocessing: Study on UO_2 Powder Removal in Electrefining Process

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Pyroprocessing technology has emerged as a viable alternative for the treatment of metal/oxide used fuel within the nuclear fuel cycle. This innovative approach involves an oxide reduction process wherein spent fuel in oxide form is placed within a cathode basket immersed in a molten $\text{LiCl-Li}_2\text{O}$ salt operating at 923 K. The chemical reduction of these oxide materials into their metallic counterparts occurs through a reaction with Li metal, which is electrochemically deposited onto the cathode. However, during process, the generation of Li_2O within the fuel basket is inevitable, and due to the limited reduction efficiency, a significant portion of rare earth oxides (REO_x) remains in their oxide state.

The presence of these impurities, specifically Li_2O and REO_x , necessitates their transfer into the electrefining system, leading to several challenges. Both Li_2O and REO_x exhibit reactivity with UCl_3 , the primary electrolyte within the electrefining system, causing a continuous reduction in UCl_3 concentration throughout the process. Furthermore, the formation of fine UO_2 powder within the salt system, resulting from chemical reactions, poses a potential long-term operational and safety concern within the electrefining process. Various techniques have been developed to address the issue of UO_2 fine particle removal from the salt, utilizing both chemical and mechanical methods. However, it is crucial that these methods do not interfere with the core pyroprocessing procedure.

This study aims to investigate the impact of Li_2O and REO_x introduced from the electrolytic reduction process on the electrefining system. Additionally, we propose a method to effectively eliminate the generated UO_2 fine powder, thereby enhancing the long-term operational stability of the electrefining process. The efficiency of this proposed solution in removing oxidized powder has been confirmed through laboratory-scale testing, and we will provide a comprehensive discussion of the detailed results.

Keywords: Pyroprocessing, UO_2 powder, Electrefining process, Lithium oxide

Comparative Assessment of Radiation Shielding in Low Water Level Spent Fuel Pools

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After the Fukushima disaster, overseas nuclear power plants have established conditions for issuing a red alert in the event of fuel damage within the spent fuel pool and they have already implemented conditions for issuing a blue alert when fuel is exposed above the water surface. In South Korean nuclear power plants, a real-time monitoring system is in place to oversee the exposure of spent fuel to the surface within the spent fuel pool. To achieve this, a water level indicator gauge is installed within the spent fuel pool, allowing for continuous real-time monitoring. This paper conducted a comparative assessment of radiation levels from water level monitoring system in two units' spent fuel pools based on the low water levels (1 feet from the storage rack), utilizing the radiation analysis code (MCNP).

Keywords: Spent fuel, Red alert, Monitoring, MCNP, Radiation shielding

Thermal Analysis of Modular Electrolytic Recovery Process Basket Assembly

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The process basket assembly is an important module in pyroprocess, because pyroprocess is a batch process, so process materials are contained in a basket assembly and transferred with the basket. The basket assembly is composed of upper and lower assembly. The lower assembly is a basket or crucible which contains process materials, and it can have electrodes. The upper assembly mainly consists of heat shields, a flange, and connectors for supplying currents to electrodes of the lower assembly.

During the electrolytic recovery process, the lower part is submerged into molten salt, whose temperature is about 500°C at least and the heat from salt is transferred to the upper assembly. And the heat affects the performance or durability of parts on the top of equipment and can raise cell temperature, which is an undesired situation. In addition, the handling equipment can pick the assembly when it is hottest, and during the transfer, the gripped part is under thermal and mechanical stresses. Because of this, the thermal effects from the heat should be required during equipment design stage.

In this study, the thermal analyses of process basket assembly were conducted for 3 cases: the steady state of the basket assembly when it submerged in molten salt, the thermal and mechanical stresses when gripped by remote handling device, and the temperature changes under natural convection. These analyses were performed using Solidworks with flow simulation package, and the results will apply to improve the thermal resistant performance of the basket assembly.

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Keywords: Thermal analysis, Process fuel basket, Electrolytic recovery equipment

Comparative Study of Dry Storage Condition Characteristics of HANA and Commercial Cladding

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In nuclear fuel development research, consideration of the back-end cycle is essential. In particular, a review of an in-reactor performance of nuclear fuel related to the various degradation phenomena that can occur during spent fuel dry storage is an important area.

The important factors affecting the degradation of zirconium-based cladding during dry storage are the cladding's hydrogen concentration and rod internal pressure after irradiation.

In this study, a preliminary analysis of the in-reactor behavior of the HANA cladding, which has been developed and is currently undergoing licensing review, was performed, and based on this result, a comparative analysis between nuclear fuel with HANA cladding and current commercial fuel under storage conditions was performed.

The results show that the rod internal pressure of nuclear fuel with HANA cladding is not significantly different from that of commercial cladding, and the hydrogen concentration in the cladding tends to reduce due to the increased corrosion resistance, so fuel integrity in a dry storage conditions is not expected to be a major problem.

Although the lack of cladding creep data under dry storage conditions, the results from the Halden research reactor test comparing in-reactor creep behavior with Zircaloy-4 showed that there is sufficient margin for degradation due to creep during storage.

Keywords: HANA cladding, Nuclear fuel, Dry storage, Hydrogen, Rod internal pressure

A Preliminary Study on TLAA for Boron Depletion of Neutron Poison in a Dry Storage System of Spent Nuclear Fuel

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International Atomic Energy Agency defines the term “Poison” as a substance used to reduce reactivity, by virtue of its high neutron absorption cross-section, in IAEA glossary. Poison material is generally used in the reactor core, but it is also used in dry storage systems to maintain the subcriticality of spent fuel. Most neutron poison materials for dry storage systems are boron-based materials such as Al-B Carbide Cermet (e.g., Boral®), Al-B Carbide MMC (e.g., METAMIC), Borated Stainless Steel, Borated Al alloy. These materials help maintain subcriticality as a part of the basket.

U.S.NRC report NUREG-2214 provides a general assessment of aging mechanisms that may impair the ability of SSCs of dry storage systems to perform their safety functions during long-term storage periods. Boron depletion is an aging mechanism of neutron poison evaluated in that report. Although that report concludes that boron depletion is not considered to be a credible aging mechanism, the report says analysis of boron depletion is needed in original design bases for providing long-term safety of DSS.

Therefore, this study aimed to simulate the composition change of neutron poison material in the KORAD-21 system during cooling time considering spent fuel that can be stored. The neutron source term of spent fuel was calculated by ORIGEN-ARP. Using that source term, neutron transport calculation for counting neutrons that reach neutron poison material was carried out by MCNP®-6.2. Then, the composition change of neutron poison material by neutron-induced reaction was simulated by FISPACT-II. The boron-10 concentration change of neutron poison material was analyzed at the end. This study is expected to be the preliminary study for the aging analysis of neutron poison material about boron depletion.

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Keywords: Neutron poison, Spent fuel, Dry storage system, Boron depletion

Revisit of Radiotoxicity as a Relative Safety Index for Radioactive Waste

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The International Atomic Energy Agency (IAEA) Safety Fundamentals No. SF-1 Safety Principle 7 states that people and the environment, present and future, must be protected against radiation risk. Therefore, it is important to evaluate the safety of radioactive waste repositories on a long-term time scale to ensure future safety. However, IAEA-TECDOC-767 states that the long-term time scale of interest means that the risk or dose to future individuals cannot be reliably predicted because it relies on assumptions. Therefore, evaluating the safety of long-term time scales should use safety indicators that are less dependent on assumptions. Radiotoxicity is one of the safety indicators that represent an inherent risk from radioactive waste. It has been mainly used to show the time required until the hazard presented by waste decreases to that of natural uranium ore and is easy to use in communication with the public.

There are several methods for calculating Radiotoxicity. Radioactivity is multiplied by a Dose Conversion Factor (DCF) to be expressed in Sv units, or radioactivity be divided into Maximum Permissible Concentration (MPC) to be expressed in m3 units as the amount of water needed to dilute the radionuclide to the permitted level. It is also often made dimensionless through comparison with reference materials like uranium ore. Radiotoxicity varies in size several times, even if it is a waste of similar origins and components, depending on the Radiological variable (e.g., Annual Limitation Intake (ALI), Dose Conversion Factor (DCF), Maximum Permissible Concentration (MPC), Activity). Therefore, this study was conducted to determine whether there was a significant difference when different radiological variables were substituted.

This study compares and analyzes their differences using various MPCs or DCFs used in each country. In addition, this study analyzes radionuclides that influence radiotoxicity with several radiological variables. This study introduces the effects of substituting different radiological variables.

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Keywords: Radiotoxicity, Safety indicator, Radioactive waste, Radiological variable

Experimental-Parametric Study on Residual Water During Vacuum Drying of Spent Nuclear Fuel Canisters

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The saturation of wet storage facilities constructed and operated within nuclear power plant sites has magnified the significance of research concerning the dry storage of spent nuclear fuel. Not only do wet storage facilities incur higher operational and maintenance costs compared to dry storage facilities, but long-term storage of metal-clad fuel assemblies submerged in aqueous tanks is deemed unsuitable. Consequently, dry storage is anticipated to gain prominence in the future. Nevertheless, it is widely acknowledged that quantitatively assessing the residual water content remains elusive even when employing the apparatus and procedures utilized in the existing dry storage processes. The presence of residual water can only be inferred from damage or structural alterations to the spent nuclear fuel during its dry storage, making precise prediction of this element crucial, as it can be a significant contributor to potential deformations and deterioration. The aforementioned challenges compound the issue of retrievability, as substantial complexities emerge when attempting to retrieve spent nuclear fuel for permanent disposal in the future. Consequently, our research team has established a laboratory-scale vacuum drying facility to investigate the sensitivity of various parameters, including canister volume, pump capacity, water surface area, and water temperature, which can exert thermohydraulic influences on residual water content. Moreover, we have conducted dimensional analysis to quantify the thermohydraulic effects of these parameters and express them as dimensionless numbers. These analytical approaches will subsequently be integrated into predictive models for residual water content, which will be further developed and validated at pilot or full-scale levels. Furthermore, our research team is actively engaged in experimental investigations aimed at fine-tuning the duration of the pressure-holding phase while optimizing the evaporation process under conditions designed to avert the formation of ice caused by abrupt temperature fluctuations. Given that the canister is constructed from acrylic material, we are able to identify, from a phenomenological perspective, the specific juncture at which the boiling phenomenon becomes manifest during the vacuum drying process.

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Keywords: Vacuum drying, Spent nuclear fuel, Dry storage system, Residual water, Heat-transfer

Drying Tests According to Pump Capacity and Canister Volume

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When storing spent fuel in a dry condition, it becomes essential to ensure that any remaining moisture bound to the canister and spent fuel is effectively removed and stored within an inert gas environment. This is crucial for preserving the integrity of the spent fuel. According to the NRC-02-07-C-006 report, it is advised to reduce pressure gradually or in incremental stages to prevent the formation of ice. In the context of vacuum drying, it is desirable to perform testing using a prototype model; however, utilizing a prototype model can be difficult due to budget constraints. To address this limitation, we designed and constructed a laboratory-scale vacuum drying apparatus. Our aim was to assess the impact of vacuum pump capacity on the drying process, as well as to evaluate the influence of canister volume on drying efficiency. The vacuum drying tests were carried out until the surface temperature of the water inside reached 0.1°C. In the tests focusing on vacuum pump capacity, vacuum pumps with capacities of 100, 200, 400, and 600 liters were employed. The outcomes of these tests indicated that smaller vacuum pump capacities resulted in increased evaporation rates but also prolonged drying times. In the case of drying tests based on canister volume, canisters with volumes of approximately 100 and 200 liters were utilized. The results revealed that larger canister volumes led to longer drying times and lower rates of evaporation. Consequently, if we were to employ an actual dry storage cask for vacuum drying the interior of the canister, it is anticipated that the process would require a substantial amount of time due to the considerably larger volume involved.

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Keywords: Vacuum drying, Spent fuel, Canister, Lab-scale, Storage cask

Introducing a Particle Size Analyzing System for Making Uranium Oxide Blocks

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Currently, the Korea Atomic Energy Research Institute is conducting research on the development of technology to reduce the disposal area for SF (Spent nuclear Fuel).

If the main radionuclides contained in SF can be separated and recovered according to their characteristics (long half-life, high mobility and high heat load) and uranium oxide which is expected to be the final residue, can be made into solids, the burden of the permanent disposal area of the SF will be greatly reduced. The waste form that end up in the repository must be verified for ease of manufacture and stability of the block. And, in order to increase the loading efficiency, a large block manufacturing technology is needed. This study describes the background of introducing PSA (Particle Size Analyzer) which is one of the necessary equipment for manufacturing UO_2 blocks using slip casting, the method of using the equipment and performance verification of the equipment using standard samples.

The particle size affects the sintering quality by the way the particles rearrange themselves during sintering. Powders of small particles are generally less free flowing and more difficult to compress, they form thin pores between the particles and sinter to higher density. In contrast, larger particle has a lower sintered density. Therefore, accurate particle size measurement and the selection of a suitable particle size are important. For this purpose, a PSA was installed in nuclear cycle experiment research center. To verify the performance of the equipment, a standard sample of $1.025\ \mu\text{m}$ was analyzed. We got an average particle size of $1.0293\ \mu\text{m}$ and standard deviation of $0.0668\ \mu\text{m}$. This value was within the uncertainty ($\pm 0.018\ \mu\text{m}$) of the sample's certificate. In the future, this equipment will measure the size of UO_2 (depleted uranium) powder and to produce large scale uranium oxide blocks.

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Keywords: Particle size analyzing, UO_2 block, Slip casting

Preliminary Review of Irradiation Tests for Neutron Absorbers in Dry Storage Systems of Spent Nuclear Fuel

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Notice of the NSSC No.2021-14 defines the term ‘Neutron Absorber’ as a material with a high neutron absorption cross section, which is used to prevent criticality during nuclear fission reactions and includes neutron absorbers as target items for manufacture inspection.

U.S.NRC report of the NUREG-2214 states that the subcriticality of spent nuclear fuel (SNF) in Dry Storage Systems (DSSs) may be maintained, in part, by the placement of neutron absorbers, or poison plates, around the fuel assemblies. This report mentions the need for Time-Limited Aging Analysis (TLAA) on depletion of Boron (^{10}B) in neutron absorbers for HI-STORM 100 and HI-STAR 100. Also, this report mentions that ^{10}B depletion occurs during neutron irradiation of neutron absorbers, but only 0.02% of the available ^{10}B is to be depleted through conservative assumptions regarding the neutron flux or accumulated fluence during irradiation, which supports the continued use of the neutron absorbers in the SNF dry storage cask even after 60 years of evaluated period.

There are several types of commercially available neutron absorbers, broadly classified into Boron Carbide Cermets (e.g., Boral®), Metal Matrix Composites (MMC) (e.g., METAMIC), Borated Stainless Steel (BSS), and Borated Al alloy.

While irradiation tests for neutron absorbers are primarily conducted during wet storage systems, there are also some prior studies available on irradiation tests for neutron absorbers during dry storage systems. For examples, there is an analysis of previous research on high-temperature irradiation test of metallic materials and identification of limitations in existing methodologies were conducted. Furthermore, an improvement plan for simulating the high-temperature irradiation damage of neutron absorbers was developed. In report published by corrosion society summarizes the evaluation results of the degradation mechanisms for Stainless Steel- and Al-based neutron absorbers used in SNF dry storage systems.

Acknowledgements

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Keywords: Spent nuclear fuel, Dry storage system, Neutron absorber, Irradiation test

Enhancing the Strength of Reticulated Foams by Infiltration Under Vacuum

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Reticulated foams have a continuous skeleton network consisting of aluminosilicates and are used for capturing gaseous cesium released from spent nuclear fuel at high temperature. It has high stability to high temperature and good capturing performance. Homogeneous cell distribution and modified surface structures are indispensable conditions for stable operation and handling. In particular, triangularly shaped holes inside the struts were generated during the pyrolysis of polyurethane sponge as a sacrificial template, which lead to limite the strength of the reticulated foam as well as cracks. However, several attempts have been focused on the increasing the strut thickness.

Here, we have prepared ceramic foams by the polyurethane sponge replication method with roller squeezing. Ceramic slurry including additives was determined with consideration of its viscous behavior. After pre-sintering, infiltration under vacuum was conducted. Metakaolin slurry was filled partially into the triangular void. As a result, the compression strength was improved by structure modification without composition change.

Keywords: Ceramic filter, Cesium, Reticulated foam filter, Vacuum infiltration, Strength

Toughening Kaolin-Based Ceramic Foams by Recoating

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One of released radioactive gases from a spent fuel is cesium (^{137}Cs) as semi-volatile fission products and reticulated ceramic foam could be used for capturing the gaseous cesium. It has three-dimensional open-pore structures and consumes cesium above 600°C to form cesium species including Cs-nepheline (CsAlSiO_4) and pollucite ($\text{CsAlSi}_2\text{O}_6$) phases. Kaolinite-based foam filter is a favorable ceramic filter because they exhibit superior capture characteristics compared to other aluminosilicate minerals and other shape filters. However, for usage in special conditions, structural limitations such broken struts must be improved.

Here, recoating by using centrifugation, followed by a pre-sintering cycle was conducted for covering the cracks and voids, resulting from the burnout of the polyurethane sponge as a sacrificial template. The slurry including additives was chosen by considering viscous behavior of slurries for the centrifugation. The microstructure and strength was improved by the recoating.

Keywords: Ceramic foam filter, Cesium, Kaolinite, Strut, Crack

Conceptual Design of Long-Term Storage Demonstration Test Technology for CANDU Spent Fuel Dry Storage Silo

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CANDU spent fuel dry storage silo has been operated since 1992 under 50 years operation license and it should prepare next step which could be transported to the centralized storage by the government or re-licensed at Wolsung site for operation extension. The former option needs a dynamic analysis and the latter option deals with steady static analysis. In any case, most important topic is the safety of spent fuels inside the baskets in both possible options.

Spent fuel integrity heavily relies on temperature effect which could impact material degradation and deformation. Therefore, it is very important to evaluate temperature inside the silo. In order to understand temperature profile, a Long-Term Storage Demonstration Test Technology is under development by using thermal analysis modeling and real-scale validation experiments.

This R&D consists of 3 major steps: simulated spent fuel rods and basket testing (called SEPA), assembled but independent 9 basket testing (called ASSEM), and modification design of silo top plug for thermocouple penetration (called DEMO). This article deals with conceptual design of SEPA and ASSEM. Conceptual design includes numbers, locations, interfaces and management of thermocouples & heaters which were produced base on thermal analysis assistance.

Conceptual design will be expanded to detail design, manufacture, experiment, and data production of SEPA & ASSEM system and the results of them will serve the critical information of DEMO design.

Acknowledgements

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Keywords: CANDU spent fuel, Dry storage, Demonstration

A Study on the Ring Tensile Properties of Hydrogenated Zircaloy-4 Cladding Specimens Subjected to Hydride Reorientation Test

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The hydride reorientation (HR) of used nuclear fuel cladding after operation affects the integrity during intermediate and disposal storage, as well as the handling processes associated with transportation and storage. In particular, during dry storage, which is an intermediate storage method, the radial hydrogen redistributes into circumferential hydrogen, increasing the embrittlement of used nuclear fuel cladding. This hydride reorientation is influenced by various key factors such as circumferential stress (hoop stress) due to internal rod pressure, maximum temperature reached, cooling rate during storage, and the concentration of precipitated hydrogen during irradiation.

To simulate long-term dry storage of used nuclear fuel, hydrogenated Zircaloy-4 cladding (CWSRA) specimens were used in hydride reorientation tests under various hoop stress conditions (70, 80, 90, and 110 MPa) for extended cooling periods (3 months, 6 months, and 12 months). After the hydride reorientation tests, the cladding's offset strain (%) was evaluated through a ring compression test, a mechanical property test encompassing both ductility and brittleness. In this study, the offset deformation of the hydride reorientation specimens was compared and evaluated through ring tensile tests.

In this study, the offset deformation values were compared and evaluated through ring tensile tests of the hydride reorientation test specimens. Hydrogen in zirconium cladding reduces ductility from a physical perspective and induces rapid plastic deformation. Generally, even in hydrogenated unirradiated cladding, it maintains a tensile strength of around 800 MPa at room temperature. However, high hydrogen content accelerates plastic deformation. In contrast, samples with radial hydrogen distribution exhibit fracture behavior in the elastic region below 500 MPa. This is attributed to the directional of radial hydrogen distribution. Specimens with a hydrogen concentration of 200 ppm fracture faster than those with hydrogen concentrations exceeding 400 ppm. This is believed to be due to the ease of reorientation of radial hydrogen in cladding with relatively low hydrogen content. Although the consistency of the test results is not ideal, ongoing research is needed to identify trends in hydride reorientation from a cladding perspective.

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Keywords: Hydride reorientation, Ring tensile test, Hydrogen concentration, Radial hydrogen

Evaluation of the Chloride-Induced Stress Corrosion Cracking on the Welding Part of Canister Using DCPD & EBSD Measurement

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One of the options for spent fuel dry storage systems is to store them in canisters using metal or concrete casks close to shore. The interaction between the austenitic stainless steel and the chloride atmosphere generated from the sea creates detrimental conditions leading to chloride induced stress corrosion cracking (CISCC) in the canister. The corrosion integrity of the canister in the concrete cask is very important because the canister is sealed and used for a long period of time. A canister made of austenitic stainless steel has several welding lines on the wall and lid, which are generated during the welding process and have high residual tensile stress. The interaction between the austenitic stainless steel and the chloride atmosphere generated from the sea creates detrimental conditions leading to chloride induced stress corrosion cracking (CISCC) in the canister. The corrosion integrity of the canister in the concrete cask is very important because the canister is sealed and used for a long period of time. In order to evaluate such soundness, an accelerated test capable of simulating the CISCC crack propagation phenomenon of the canister weld is required.

In this study, a test device for performing the CISCC simulation test was constructed using the DCPD device. The direct current potential drop (DCPD) technique is a widely accepted method of monitoring crack initiation and growth in controlled laboratory tests. Total 10 types of test specimens with varying welds, base metal, salinity and stress were selected and a sealed chamber with DCPD test apparatus were designed and constructed to evaluate them. The chamber for CISCC simulation was manufactured as a sealed with a solution containing 10% MgCl_2 . A 1/2 CT specimen with pre-cracked pre-cracks was loaded into the prepared container, and gauze was attached from the bottom for smooth delivery to the specimen to facilitate penetration of chloride. After the test, the measured DCPD data were correlated with Electron Back scattered Diffraction (EBSD) data.

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Keywords: Spent nuclear fuel canister, Austenitic stainless steel, Chloride induced stress corrosion cracking tester

Development of Inspection System for CANDU Spent Fuel Dry Storage Facility

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Spent fuel from the Wolseong CANDU reactor has been stored in above-ground dry storage canisters. The operating license for the dry storage canisters needs to be extended. It is desired to perform in-situ inspections of the canisters and the fuel baskets to verify their condition is suitable for retrieval (if necessary), and that the temperature within the fuel baskets is as predicted in the canister's design basis. Wolseong concrete dry storage canisters (silo) are 6.5 m high, 3 m in outside diameter, and have shielding comprised of 0.96 m of concrete and 9.5 mm (3/8") of steel liner. The storage configuration is such that a number of fuel bundles are placed inside a cylindrical steel container known as a Fuel Basket. The canisters hold up to 9 baskets each that are 304L stainless steel, 1 m in diameter, 0.56 m in height, and hold 60 fuel bundles each. The inspection equipment to be deployed for the inspection of the canisters is made up of four primary sub-assemblies: the work platform, the delivery system, the camera tool, and the control system. Therefore, this paper describes the development of an internal inspection system for silos, a dry storage facility in Wolseong.

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Keywords: CANDU spent fuel, Silo, Inspection system

Solvent Extraction of U From the Aqueous Solution and Detection of Radioactivity of α and γ Emitting Species

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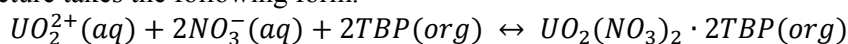
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To address the pressing societal concern in Korea, characterized by the imminent saturation of spent nuclear fuel storage, this study was undertaken to validate the fundamental reprocessing process capable of substantially mitigating the accumulation of spent nuclear fuel. Reprocessing is divided into dry processing (pyro-processing) and wet reprocessing (PUREX). Within this context, the primary focus of this research is to elucidate the foundational principles of PUREX (Plutonium Uranium Redox Extraction). Specifically, the central objective is to elucidate the interaction between uranium (U) and plutonium (Pu) utilizing an organic phase consisting of tributyl phosphate (TBP) and dodecane. The objective was to comprehensively understand the role of HNO_3 in the PUREX (Plutonium Uranium Redox Extraction) process by subjecting organic phases mixed with TBP-dodecane to various HNO_3 concentrations (0.1 M, 1.0 M, 5.0 M). Subsequently, the introduction of Strontium (Sr-85) and Europium (Eu-152) stock solutions was carried out to simulate the presence of fission products typically contented in the spent nuclear fuel. When the operation proceeds, the complex structure takes the following form.



Subsequently, separate samples were gathered from both the organic and aqueous phases for the quantification of gamma-rays and alpha particles. Alpha particle measurements were conducted utilizing the Liquid Scintillation Counter (LSC) system, while gamma-ray measurements were carried out using the High-Purity Germanium Detector (HPGe). The distribution ratio for U, Eu (Eu-152), and Sr (Sr-84) was ascertained by quantifying their activity through LSC and HPGe.

Through the experiments conducted within this program, we have gained a comprehensive understanding of the selective solvent extraction of actinides. Specifically, uranium has been effectively separated from the aqueous phase into the organic phase using a combination of tributyl phosphate (TBP) and dodecane. Subsequently, samples containing U(VI), Eu(III), and Sr(II) underwent thorough analysis utilizing LSC and HPGe detectors. Our radiation measurements have firmly established that the concentration of nitric acid enhances the selective separation of uranium within the process.

Acknowledgements

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Keywords: PUREX (plutonium uranium redox extration), Uranium extraction, TBP, Distribution coefficient

A Preliminary Criticality Analysis of Fuel Storage System for Accident Tolerant Fuel

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Since the Fukushima nuclear accident in 2011, the development of accident tolerant fuel (ATF) has been actively pursued as an alternative to improve the safety of nuclear power plants. In addition, nuclear power plants containing ATF have recently been included as green energy in the 2022 EU taxonomy bill, receiving a lot of attention. Many countries are considering increasing ^{235}U enrichment from 5 to 10 ^{235}U % for higher burnup and long cycle operation with ATF improving safety. To utilize ATF, the applicability of fuel storage systems such as new fuel storage vault, Region 1, and Region 2 must be determined. The purpose of this paper is to confirm the applicability of applying ATF, which is being developed in Korea, to the nuclear fuel storage system of Korean nuclear power plants. The nuclear power plant model used in the analysis is APR-1400, a representative Korean nuclear power plant model, and ATF model used in the analysis is Mo microcell UO_2 pellet with CrAl coating, which is being developed in Korea. MCNP 6.2 has been used for multiplication factor calculations, and the TRITON/NEWT and ORIGEN-S modules of the SCALE code have been used for depletion calculations. From the analysis results, solutions and additional analysis would be necessary to satisfy criticality regulatory requirements to utilize ATF with increased enrichment.

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Keywords: Accident tolerant fuel, Spent fuel pool, New fuel storage vault, Low-enriched uranium+

Comparative Study on Background Signal of the Optical Fiber Cable of the Existing and New Equipment for CANDU-Type Spent Fuel Verification

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During PIV (Physical Inventory Verification), the IAEA has been inspecting the CANDU-Type spent fuels using an optical fiber-based scintillation detector. KINAC has developed a new verification instrument to deal with problems of the existing one such as low sensitivity, heavy and large dimension, and inconvenience-in-use. Our previous studies focused on how to develop the new instrument and had not included its performance tests. Field tests were carried out recently at Wolsung unit 4 to evaluate performance of the existing and new instruments. The objective of this paper is to discuss background noise produced in the optical fiber signal cable itself.

The verification equipment for the CANDU-type Heavy Water Reactor spent fuels uses a scintillation detector to bond a scintillation material to the end of an optical signal cable. At this time, the radiation signal obtained by a data acquisition system is the signal generated from the scintillator (p-terphenyl organic scintillator) and the optical signal cable ; The signal produced in the optical cable itself is background noise to degrade the spent fuel verification equipment. To characterize the background radiation noise, the spent fuel bundles at Wolsung Unit 4 were measured using the optical fiber cable without the radiation scintillator. This signal is generated by reaction of the optical cable and the radiation emitted from the spent fuel.

From experimental results, it was observed that the background noise signal of the optical cable increased as the optical cable went down in the downward direction, because the cable length irradiated by the radiation increased with the optical cable area in the spent fuel storage pool. Difference in the background noise signal was dependent on the location of the vertical direction and the signal of the new optical cable was up to about 5 times higher than that of the existing cable. While, the new cable has the cross-section area about 3.2 times larger than the old cable.

Our past studies showed that total signal amplitude – sum of signals generated from the scintillator and optical fiber - of the new verification instrument was at least about 15 times greater than that of the existing one. Considering the total signal and background noise signal, from this measured results, it was confirmed that the scintillator characteristics – in particular, light output and decay time – has a dominant impact on the signal sensitivity of the newly developed instrument. More details will be discussed at the conference.

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Keywords: Nuclear safeguards, Spent fuel bundle, CANDU, Verification, Scintillation detector, Optical fiber

Comparison of Thermal Analysis With CFD Code and COBRA-SFS for Dry Cask Simulator

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Spent nuclear fuels should be safely stored until being disposed and dry storage system is predominantly used to retain the fuels. Thermal analysis to estimate temperatures of spent nuclear fuel and the storage system should be performed to evaluate whether the temperatures exceed safety limit. Recently, thermal hydraulic analysis with CFD codes is widely used to investigate the temperature of spent nuclear fuel in dry storage. COBRA-SFS is a legacy code based on sub-channel analysis code, and its fidelity is verified for evaluating the thermal analysis for licensing a dry cask system.

Herein, thermal analysis result based on CFD and COBRA-SFS codes is compared and the Dry Cask Simulator (DCS) is assessed as a benchmark experiment in this study. Extended Storage Collaborating Program (ESCP) led by the Electric Power Research Institute (EPRI) is organized to address the degradation effects of spent nuclear fuel during long-term dry storage, and DCS is the first phase of the program. The dry storage system, containing a single BWR assembly in a canister, was designed to produce validation-quality data for thermal analysis model.

ANSYS FLUENT was used to simulate DCS. Simulations were conducted in various decay heat and helium pressure inside the canister. In realistic conditions of decay heat and helium pressure of actual dry cask system, CFD and COBRA-SFS analysis results gave good agreement with experimental measurement. Peak temperatures of channel can, basket, canister and shell predicted by CFD simulation also showed good prediction and the discrepancies were less than 7 K while measurements uncertainty was 7 K. In high decay heat and high pressure condition, however, CFD and COBRA-SFS underestimated peak cladding temperature than experimental results.

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Keywords: Dry storage, Dry cask simulator, ESCP, CFD validation, Peak cladding temperature

Heat Treatment and TG-DTA Analysis of Uranyl Nitrate and Uranium Oxides

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The solid-state chemistry of uranium is essential to the nuclear fuel cycle. Uranyl nitrate is a key compound that is produced at various stages of the nuclear fuel cycle, both in front-end and back-end cycles. It is typically formed by dissolving spent nuclear fuel in nitric acid or through a wet conversion process for the preparation of UF_6 . Additionally, uranium oxides are a primary consideration in the nuclear fuel cycle because they are the most commonly used nuclear fuel in commercial nuclear reactors. Therefore, it is crucial to understand the oxidation and thermal behavior of uranium oxides and uranyl nitrates.

Under the ‘2023 Nuclear Global Researcher Training Program for the Back-end Nuclear Fuel Cycle,’ supported by KONICOF, several experiments were conducted at IMRAM (Institute of Multidisciplinary Research for Advanced Materials) at Tohoku University. First, the recovery ratio of uranium was analyzed during the synthesis of uranyl nitrate by dissolving the actual radioisotope, U_3O_8 , in a nitric acid solution. Second, thermogravimetric-differential thermal analysis (TG-DTA) of uranyl nitrate ($UO_2(NO_3)_2$) and hyper-stoichiometric uranium dioxide (UO_{2+x}) was performed. The enthalpy change was discussed to confirm the mechanism of thermal decomposition of uranyl nitrate under heating conditions and to determine the chemical hydrate form of uranyl nitrate. In the case of UO_{2+x} , the value of ‘x’ was determined through the calculation of weight change data, and the initial form was verified using the phase diagram for the U-O system. Finally, the formation of a few UO_{2+x} compounds was observed with heat treatment of uranyl nitrate and uranium dioxide at different temperature intervals (450°C-600°C). As a result of these studies, a deeper understanding of the thermal and chemical behavior of uranium compounds was achieved. This knowledge is vital for improving the efficiency and safety of nuclear fuel cycle processes and contributes to advancements in nuclear science and technology.

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Keywords: Enthalpy change, Heat treatment, Hyper-stoichiometric uranium dioxide, TG-DTA, Uranyl nitrate

An Experimental Investigation of the Correlation Between Lid Displacement and Leakage Rate of Transport Cask and Application of CFD

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Considering the domestic situation where all nuclear power plants are located on seaside, the interim storage site is also likely to be located on coastal site. Maritime transportation is inevitable and its risk assessment is very important for safety. Currently, there is no independently developed maritime transportation risk assessment code in Korea, and no research has been conducted to evaluate the release of radioactive waste due to the immersion of transport cask.

Previous studies show that the release rate of radionuclides contained in a submerged transport cask is significantly affected by the area of flow path generated at the breached containment boundary. Due to the robustness of a cask, the breach is the most likely generated between the lid and body of cask. CRIEPI investigated the effect of cask containment on the release rate of radioactive contents into the ocean and proposed a procedure to calculate the release rate considering the so-called barrier effect. However, the contribution of O-ring on the release rate was not considered in the work.

In this study, test and analysis is performed to determine the equivalent flow path gap considering the influence of O-rings. These results will be implemented in the computational model to assess sea water flow through a breached containment boundary using CFD techniques to assess radionuclide release rates. To evaluate the release rate as a function of lid displacement, a small containment vessel is engineered and a metal O-ring of the Helicoflex HN type is installed, which is the most commonly used one in transport and storage casks. The lid of containment vessel is displaced in vertical and horizontal direction and the release rate of the vessel was quantified using the helium leak test and the pressure drop test. Through this work, the relationship between the vertical opening displacement and horizontal sliding displacement of the cask lid and the actual flow path area created is established. This will be implemented in the CFD model for flow rate calculation from a submerged transport cask in the deep sea. In addition, the compression of the O-ring causes very small gaps, such as capillaries. In these cases, Poiseuille's law is used to calculate the capillary flow rate.

Keywords: CFD, Release rate evaluating, Transport cask, Barrier effect model

Dynamic Material Testing of Concrete Cores From HANARO

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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, dynamic material testing was conducted on concrete cores extracted from HANARO exterior wall during seismic reinforcement construction. HANARO was constructed at the Korea Atomic Energy Research Institute in 1995, following strict nuclear quality assurance standards.

In order to conduct the dynamic material testing of the extracted concrete cores, self-disposal had to be performed because the concrete cores were extracted and stored in a radiation controlled area. A self-disposal application was prepared and submitted based on the radionuclide analysis results, and it was finally approved in April 2023. Then, a test was performed by processing a specimen for dynamic property testing using a self-disposed concrete core. The concrete cores were processed to create specimens for dynamic material testing and the dynamic material testing was performed to obtain stress-strain diagrams according to the strain rate.

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Keywords: Dry cask storage system, Interim storage facility, Dynamic material testing, Self-disposal

Simulating Evaluation for Efficient Design and Manufacture of Spent Nuclear Fuel Burnup Detector Using MCNP

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For efficient design and manufacture of PWR spent fuel burnup detector, data simulated with various condition of spent fuel in the NPP storage pool is required. In this paper, to derive performance requirements of spent fuel burnup detector for neutron flux and dose rates were evaluated at various distances from CE16 and WH17 types of fuel, representatively.

The evaluation was performed by the following steps. First, the specifications of the spent fuel, such as enrichment, burnup, cooling time, and fuel type, were analyzed to find the conditions that emit maximum radioactivity.

Second, gamma and neutron source terms of spent fuel were analyzed. The gamma source terms by actinides and fission products and neutron source terms by spontaneous and (α , n) reactions were calculated by SCALE6 ORIGAMI module.

Third, simulation input data and model were applied to the evaluation. The material composition and dose conversion factor were referred as PNNL-15870 and ICRP-74 data, respectively and dose rates were displayed with the MCNP output data. It was assumed that there was only one fuel modeled by MCNP 6.2 code in pool. The evaluation positions for each distance were selected as 5 cm, 10 cm, 25 cm, 50 cm, and 1 m apart from the side of fuel, respectively.

Fourth, neutron flux and dose rates were evaluated at distance from each fuel type by MCNP 6.2 code. For WH 17 types with a 50 GWd/MTU burnup from 5 cm distance close to fuel, the maximum neutron flux, gamma dose rates and neutron dose rates are evaluated as 1.01×10^5 neutrons/sec, 1.41×10^5 mSv/hr and 1.61×10^1 mSv/hr, respectively. The flux and dose rate of WH type were evaluated to be larger than those of CE type by difference in number of fuel rods. The relative error for result was less than 3~7% on average secured the reliability.

It is expected that the simulated data in this paper could contribute to accumulate the basic data required to derive performance requirements of spent fuel burnup detector.

Acknowledgements

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Keywords: Spent fuel burnup detector, Neutron flux, Dose rates, ORIGAMI, MCNP

Proposal of Post-Processing Method for Ring Tensile Test Results for Relative Comparison of Young's Modulus of Materials

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Ring Tensile Test (RTT) is mainly performed for comparing tensile strength and total strain between nuclear fuel cladding specimens under various initial conditions. Through RTT, the load-displacement (F-D) curve obtained from the uniaxial tensile test can also be obtained. However, the Young's modulus estimated from the gradient of the straight portion is much lower than general value of materials. The reasons include tensile machine compliance, slack in the fixtures, or elastic deformation of the fixtures and the tooling. Another reason is that the bending of the test part in the ring is stretched with two pieces of tools. Although the absolute value of the Young's modulus is smaller than the actual value, it is applicable to calculate the ratio of the Young's moduli of different materials, that is, the relative value. The Young's modulus, or slope of the linear section, varies slightly depending on which location data is used and how much data is included. In order to obtain a more accurate ratio of Young's moduli between materials using the RTT results, a post-processing method for the ring tensile test results that can prevent such human errors is proposed as follows.

First, the slope of the linear section is obtained using the displacement and load when the load increase is the largest and the displacement and load of the position that is 95% of the maximum load increase. To replace the section where the ring-shaped specimen is stretched at the beginning of the F-D curve, a straight line equal to the slope of the linear section is drawn to the displacement axis from the position of maximum load increase and moved to the origin to obtain the final F-D curve for a RTT. Lastly, the yield stress uses the stress at the point where the 0.2% offset straight line and the F-D curve meet as suggested in the ASTM E8/E8M-11 "Standard test methods for tensile testing of metallic materials".

RTT results post-processing method was coded using FORTRAN language so that it could be performed automatically. In addition, sensitivity analysis of the included data range on the Young's modulus was performed by using the included data range as 90%, 85%, and 80% of the maximum load increase.

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Keywords: Ring tensile test (RTT), Post-processing method, Relative comparison of Young's modulus, Post-processing automation

Segregation of the Precipitates and Salt Ingot From Tapered Crucible for Recovery of the Metal Phosphate Precipitates

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In KAERI's previous phosphate precipitation tests, the dispersed powder of lithium phosphate (Li_3PO_4) as a precipitation agent reacted with various metal chlorides in a simulated LiCl-KCl molten salt. The reaction of metal chlorides composed of actinides such as uranium and three rare earths (Nd, Ce and La) with lithium phosphate is a solid-liquid reaction. A phosphorylation reaction rate is very fast and the metal phosphates as a reaction product precipitated on the bottom of the molten salt crucible. One of the recovery methods of the metal phosphate precipitates is segregation the lower part (precipitates) of the salt ingot using the various cutting tools. Recently, a new phosphorylation experiment using lithium phosphate ingots carried out in order to collect the metal phosphate precipitates into a small recovering vessel, and the test result of this new method was feasible. However, the reaction rate of test using lithium phosphate ingot is extremely slower than that of test using lithium phosphate powder. In this study, the precipitation reactor design (a tapered crucible with polished inner surface) used for phosphorylation reaction showed that the salt ingot with metal phosphate precipitates could be detached from a tapered stainless steel crucible. We propose that the recovery of precipitates from a salt ingot is possible by introducing a dividing plate structure into a molten salt and by positioning it at the interface between salt and precipitated metal phosphate.

Acknowledgements

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Keywords: Uranium, Rare earths, Phosphorylation, Precipitation, Salt ingot, Tapered crucible

Rolling Test Simulating Sea Transport of Spent Nuclear Fuel

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Currently, 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 is progressing in Korea by the present authors. Road transportation tests using surrogate spent nuclear fuel were performed in September, 2020 using a test model of KORAD-21 transportation cask and sea transportation tests were conducted from September 30 to October 4, 2021. Finally, the shake table tests and rolling test were conducted from October 31 to November 2, 2022.

As a result of the sea transportation test data analysis, an impact load resulting from the collision of objects was measured on fuel rods of a surrogate spent nuclear fuel assemblies during the rolling test was observed. Excessive rolling motion occurred on the ship during the rolling test, causing the surrogate spent nuclear fuel assemblies to slip and collide with the canister. To analyze under which conditions such impact loads occur and whether this event is possible under normal conditions of transport of spent nuclear fuel, a test was designed to simulate the rolling test in sea transportation and was performed.

The rolling test was conducted on ACE7 and PLUS7 assemblies, respectively, varying the rolling angle and rolling frequency to determine at which angles and frequencies the assemblies experienced slippage. According to the test results, slippage of the used nuclear fuel assemblies can occur due to rolling motion at angles of approximately 14° or higher, leading to the possibility of generating impact loads. It was observed that the rolling angle is a more major factor for slippage than the rolling frequency. This exceeds the conditions under which a vessel can be permitted to depart for coastal navigation, thus it is considered to deviate from the normal conditions of transport of spent nuclear fuel. Therefore, it is not necessary to consider such loads for evaluating the integrity of spent nuclear fuel during normal transportation conditions.

Acknowledgements

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Keywords: Normal conditions of transport, Spent nuclear fuel, Sea transport, Rolling test, Structural integrity

Design and Fabrication of Automated Device for Salt Sample Analysis

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As part of strengthening pyro safety measures, the Korea Atomic Energy Research Institute is developing LIBS (Laser-Induced Breakdown Spectroscopy) application technology to analyze molten salt components in electrolytic recovery device in real time. LIBS performs qualitative and quantitative analysis by analyzing the spectrum of energy emitted by atomizing and ionizing elements on the surface of a salt sample with a high-focused laser. Since salt easily corrodes metal, it must be managed in an environment with a dew point of -40°C or lower. In this study, we designed and manufactured a device that places a rod-type sampling stick on a mounting base, automatically moves it to the optimal measurement position for LIBS, and retrieves the sample. Its characteristics are as follows. First, LIBS is stationary and does not move. Second, the sample stick is placed on a mounting base and can rotate 360 degrees. Third, according to the command, the sample stick automatically moves to the optimal measurement position of LIBS with three degrees of freedom (X, Y, Z). Fourth, the salt attached to the sampling stick is recovered for chemical analysis by driving the gripper mounted at the bottom of the Z axis, Z axis, and rotation axis (R). The X, Y, and Z movement distances of this device are each 100 mm, rotation is 360 degrees, grip stroke is 50 mm, and position accuracy is $\pm 20\text{ }\mu\text{m}$. Once the performance test of the automated salt sample analysis device is completed, it will be installed in a dry room with a dew point of -40°C or lower. Samples will be collected remotely in connection with the electrolytic recovery device and gantry robot built in the dry room. We plan to conduct experiments to seat the sample stick. Ultimately, we plan to conduct comprehensive experiments in conjunction with LIBS.

Acknowledgements

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Keywords: LIBS, Molten salt, Sampling stick, Electrolytic recovery device, Automated salt sample analysis device

A Study of Fracture Behavior of Spent Fuel in Long-Term Dry Storage

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To investigate the mechanical integrity of spent nuclear fuel, the failure behavior of the cladding tube was examined under accident conditions. According to the SNL report, the failure behavior of cladding can be broadly classified into two types. The first is failure due to bending load caused by falling. The second is failure due to pinch load caused by space grid.

In this study, mechanical integrity was evaluated through the stress intensity factor applied to the crack in failure behavior due to bending load. Since the exact value of the impact load due to fall was unknown, the load was applied by increasing the value up to 200 G in 20 G increments. The size of the crack is an important input variable, and 300 μm was given by referring to the EPRI report, and the elastic modulus, a material property that determines the stress field, was given 75.22 GPa by referring to the FRAPCON code. Since the relationship between the direction of stress and the direction of the crack is also a major variable, simulations were conducted for both cracks perpendicular to and parallel to the stress direction.

It was confirmed that at a load of 200 G, when the crack was parallel to the stress direction, stress concentration did not occur and had a very low stress intensity factor $0.01 \text{ MPa}\sqrt{\text{m}}$. When perpendicular to the direction of stress, the stress intensity factor showed a value of $1 \text{ MPa}\sqrt{\text{m}}$. However, considering that the critical value of the stress intensity factor due to hydride is $5 \text{ MPa}\sqrt{\text{m}}$, it can be seen that perpendicular result also ensures the mechanical integrity of the cladding.

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Keywords: Spent fuel, Hydride, Reorientation, Irradiation, Degradation

An Experimental Verification Study on the Heat Flow Analysis Results of a Tube Furnace

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In general, systems are developed by repeatedly performing the processes of design, analysis, manufacturing, and performance testing. In particular, systems with temperature, pressure, and flow rate often utilize computational fluid dynamics tools at the design stage. In this paper, we aim to verify the reliability of the analysis results of Solidworks Flow Simulation, which is widely used in heat flow analysis at the design stage. A tube furnace was manufactured, various experiments were performed, and a study was conducted to compare the analysis results. The details of the experiment are as follows. First, an experiment was conducted in which the heater was heated to 900°C without insulating the exposed part of the tube. The detailed contents of the experiment are as follows; - Heating heater and measuring temperature without supplying flow inside the tube, - Tube flow supply (25°C, 15 lpm air) and heater heating/temperature measurement. Second, an experiment was performed in which the exposed part of the tube was insulated (thickness 50 mm) and the heater was heated to 900°C. The detailed contents of the experiment are as follows; - Insulate the outside of the tube except for the flanges at both ends of the tube, and heat the heater and measure the temperature without supplying flow inside the tube. - Insulate the outside of the tube except for the flanges at both ends of the tube, supply flow rate inside the tube (25°C, 15 lpm air) and measure heater heating/temperature. - Insulate the flange of the flow supply section, heat the heater and measure temperature without supplying flow inside the tube. - Insulate the flange of the flow supply section, heat the supply air (277°C, 15 lpm) and measure the temperature using a heating gun without heating the heater. - Insulate the flange of the flow supply section, supply heated air (277°C, 15 lpm) and measure heater heating/temperature. - Insulate the flange of the flow supply section and measure temperature according to heater heating (900°C) and supply temperature (25°C, 277°C 15 lpm). The following results were derived from the experimental and analysis results. - When the exposed part of the tube is insulated, the temperature inside the tube increases and the steady-state power decreases compared to non-insulated. - In areas with insulation, the temperature error between experiment and analysis results is not large. - When flow rate is supplied, there is a large temperature error in experiment and analysis results. - The temperature change after the center of the heater is not large for a temperature change of 15 lpm flow rate. From these results, it can be seen that Solidworks Flow Simulation has a significant difference from the experimental results when there is a flow rate in the tube. This was thought to be because the flow rate acts as a disturbance, and this cannot be sufficiently accounted for in the analysis. In the future, we plan to check whether there is a way to solve this problem.

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Keywords: Tube furnace, Solidworks flow simulation, Insulate, Heater, Heating gun, Flow rate

Calculation of Decay Heat in SNF Using the Latest ANSI/ANS-5.1 Method

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In the 3rd revision of NUREG-0800, which was revised in 2007, the calculation method for decay heat in the design of the Ultimate Heat Sink (UHS) for a pressurized water reactor is recommended to be based on the ANSI/ANS-5.1 method. This method employs a more complex decay heat calculation formula compared to the one introduced in Branch Technical Position ASB 9-2, which was presented in the 2nd revision. While most of the variables for decay heat calculation in ANSI/ANS-5.1 can be inferred from the methods outlined in the appendices, determining the fractions of fission products is not straightforward despite their significant impact on the results.

When reviewing documents that evaluate decay heat using the ANSI/ANS-5.1 method, it is observed that they often adopt a conservative approach by assuming that the fraction of the most influential fission product is 100%. In this study, the fractions of each fission product presented in LLNL's 2016 report were used to calculate decay heat, and the results were compared with the ASB 9-2 method and ORIGEN code results. The comparison showed that ANS 5.1 tends to yield higher decay heat values than ANS 9-2, particularly at the reference time of 1M seconds, while ORIGEN-ARP generally produced lower values. Therefore, it is concluded that even when using the ANSI/ANS-5.1 method with the fractions of each fission product for decay heat calculations in spent nuclear fuel wet or dry storage facility assessments, it provides a sufficiently conservative thermal evaluation.

Keywords: Spent nuclear fuel, Decay heat, ANSI/ANS-5.1, ORIGEN, ASB 9-2

Experiments of the Dry Mock-up System and XR (eXtended Reality) Approaches for the Pyroprocess Automation

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This paper describes the development and operation of an autonomous robotic system designed for pyroprocess automation. The unique challenges of pyroprocess automation, such as the need for a highly dry atmosphere to handle materials like chloride, are addressed through this system. For the experiments, a specialized dehumidifier and dry mock-up facility were designed to produce dry air condition. Performances of dry air conditioning for the various simulated situations were evaluated, including assessing worker access within a mock-up to determine the system's feasibility. To enable automation, containers used for processing materials were modified to fit the gripper system of the gantry robot. The loading and unloading of materials in each equipment were automated to connect them with the robotic system. This gantry robot primarily utilized macro motions to approach waypoints containing process materials, reducing the need for precise approach motions. Its tapered jaw design allowed it to grip target objects even with imperfect positioning. The robot's motions were programmed initially using a robot simulator for positioning and motion planning, and real-world accuracy was tested in a dry mock-up facility using the OPC platform. Finally, the paper discusses the potential application of XR (eXtended Reality) technology in this context, which could enhance the robot's operation and provide valuable insights into the automation process. Further analysis of XR technology's feasibility and benefits for this specific pyroprocess automation system are presented.

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Keywords: Dry room, Gantry robot, Pyroprocess, Process automation, XR (eXtended Reality)

Development of Criteria Evaluation System of Leakage Rate Test for Spent Fuel Dry Storage Systems

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It is very important that the confinement of a spent fuel storage systems is maintained because if the confinement is damaged, the gaseous radioactive material inside the storage cask can leak out and have a radiological impact on the surrounding public. For this reason, leakage rate tests using helium are required for certificate of compliance (CoC) and fabrication inspections of spent fuel storage cask. For transport cask, the allowable leakage rate can be calculated according to the standardized scenario presented by the IAEA. However, for storage cask, the allowable leakage rate is determined by the canister, facility, and site specific information, so it is difficult to establish a standardized leakage rate criterion. Therefore, this study aims to establish a system that can derive system-specific leakage test criteria that can be used for leakage test of actual storage systems.

First, the variables that can affect the allowable leakage rate for normal and accident conditions were derived. Unlike transportation systems, for storage systems, the dose from the shielding analysis and the dose from the confinement analysis are summed up to determine whether the dose standard is satisfied, and even the dose from the existing nuclear facilities is summed up during normal operation condition. For this reason, the target dose is used as an input variable when calculating the allowable leakage rate for the storage system. In addition, the main variables are the distance from the boundary of the exclusive area, the number of cask, the inventory of nuclide material in the cask, the free volume, and the internal and external pressure. Utilizing domestic and US NRC guidelines, we derived basic recommended values for the selected variables. The GASPAR-II computer code that can evaluate the dose to the public under normal operating conditions was utilized. Using the above variables, the allowable leakage rate is calculated and converted to the allowable criteria for helium leakage rate test. The developed system was used to calculate the allowable leakage rate for normal and accident conditions for a hypothetical storage system.

The leakage rate criteria calculation system developed in this study can be useful for CoC and fabrication inspections of storage systems in the future, and a GUI-based program will be built for user convenience.

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Keywords: Spent fuel, Dry storage system, Confinement evaluation, Leakage rate test, Radiation dose

Research on the Concept of Nuclide Transport in the Disposal Option System

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Nuclear power is responsible for a large portion of electricity generation worldwide, and various studies are underway, including the design of permanent deep geological disposal facilities to safely isolate spent nuclear fuel generated as a result. However, through the gradual development of drilling technology, various disposal option concepts are being studied in addition to deep geological disposal, which is considered the safest in the world.

So other efforts are also being made to reduce the disposal area and achieve economic feasibility, which requires procedures to appropriately match the waste forms generated from separation process of spent nuclear fuel with disposal option systems according to their characteristics. And safety issue of individual disposal options is performed through comparison of nuclide transport.

This study briefly introduces the pre-disposal nuclide management process and waste forms, and also introduces the characteristics of potential disposal options other than deep geological disposal. And environmental conditions and possible pathways for nuclide migration are reviewed to establish transport scenarios for each disposal option. As such, under this comprehensive understanding, this study finally seeks to explore various management methods for high-level radioactive waste to reduce the environmental burden.

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Keywords: Nuclide transport, Disposal option system, Pre-management nuclide process, Waste forms

Mass Flow Analysis of Radionuclide Management Process Scenarios

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Nuclear power generation is expected to be enlarged for domestic electricity supply based on the 10th Basic Plan of Long-Term Electricity Supply and Demand. However, the issues on the disposal of spent nuclear fuel or high-level radioactive waste has not been solved. KBS-3 concept of the deep geological disposal and pyroprocessing has been investigated as options for disposal and treatment way of spent nuclear fuel. In other way, the radionuclide management process with 6 scenarios are devised combining chlorination treatment and alternative disposal methods for the efficient disposal of spent nuclear fuel. Various scenarios will be considered and comprehensively optimized by evaluation on many aspects, such as waste quantity, radiotoxicity, economy and so on.

Level 0 to 4 were identified with the specialized nuclide groups: Level 0 (NFBC, Hull), Level 1 (Long-lived, volatile nuclides), Level 2 (High heat emitting nuclides), Level 3 (TRU/RE), Level 4 (U). The 6 options (Op.1 to 6) were proposed with the differences between scenarios, for examples, phase types of wastes, the isolated nuclide groups, chlorination process sequences. Op.1 adopts Level 0 and 1 to separate I, Tc, Se, C, Cs nuclides which are major concerns for long-term disposal through heat treatment. The rest of spent nuclear fuel will be disposed as oxide form itself. Op.2 contains Sr separation process using chlorination by $MgCl_2$ and precipitation by K_2CO_3 to alleviate the burden of heat after heat treatment process. U/TRU/RE will be remained and disposed in oxide form. Op.3 is set to pyroprocessing as reference method, but residual TRU/RE chlorides after electrorefining will be recovered as precipitates by K_3PO_4 . Op.4 introduces NH_4Cl to chlorinate TRU/RE from oxides after Op.2 applied and precipitates them. TRU/RE/Sr will be simultaneously chlorinated by NH_4Cl without $MgCl_2$ in Op.5. Then, chlorinated Sr and TRU/RE groups will be separated by post-chlorination process for disposal. But, chlorinated Sr and TRU/RE are designed not to be divided in disposal steps in Op.6.

In this study, the mass flow analysis of radionuclide management process scenarios with updated process variables are performed. The amount and composition of wastes by types will be addressed in detail.

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Keywords: Mass flow, Spent nuclear fuel, Radionuclide management process, Chlorination, Precipitation

Development of Restoration Method for KOFA Spent Fuel Assembly

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Various types of spent fuel assembly in nuclear power plants have been transported to a post irradiation examination facility (PIEF) in KAERI to examine the mechanical and chemical properties of fuel and cladding. Once the fuel assembly arrive at PIEF, it is dismantled in a pool area to extract the fuel rods. Dismantling of the fuel assembly is performed by cutting the top nozzle. Currently, couple of dismantled assemblies have been stored in a storage pool without the top nozzle in PIEF. These assemblies cannot be handled directly using a gantry crane in the pool, and thus are contained in a special basket to handle.

In this research, we developed a restoration method for a dismantled spent fuel assembly, especially for 16×16 Korea Optimized Fuel Assembly (KOFA). After reviewing the original design document and reports of KOFA, two tools are devised; an assembly tool and a tightening tool for a bolt. Since the top nozzle and dismantled KOFA can be re-assembled using a bolt, we follow the original design, size, and materials of the previously used bolt. The bolt to restore the top nozzle of KOFA is made of 321 stainless steel and has a design that fits the guideline of DIN 13-21 international standard. Our procedure can potentially be used to restore and repair the dismantled spent fuel assembly.

Keywords: KOFA, Spent fuel assembly, Top nozzle, Restoration

Development of Dummy Rod Insertion Tool for Fuel Rod Extracted Assembly

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Post Irradiation Examination Facility (PIEF) is a test facility for nuclear fuel research and development and performance evaluation. From the past to the present, assemblies and fuel rods have been transported from nuclear power plants (NPP) several times, and various destructive and non-destructive tests have been performed. Among these, in the case of the 14×14 Westinghouse STD assemblies that are transported as a whole assembly, the top nozzle is connected to the guide tube by welding. Therefore, the fuel rods could not be removed from the assembly at the NPP, so the assemblies were transported to PIEF as is. Then, after cutting between the top nozzle and the guide tube in the pool, and the fuel rods were extracted and tested. In order to transport the assembly in the future, it is necessary to maintain stability by inserting the dummy rod into the unit cell from which the fuel rod is extracted. However, since the length of the dummy rod is almost 4 m and the diameter is about 10 mm, the dummy rod often bends while passing through the dimple spring of the grid. Additionally, when dummy rods are inserted into unit cells that are continuously empty after the fuel rods are extracted, there may be cases where the dummy rods are not inserted into the desired unit cell but are bent and incorrectly inserted into the next unit cell. The moment the dummy rods are inserted into the dimple spring of grid, a load is applied to the dummy rod due to the tension of the spring. If it can be inserted while offsetting the load, the work can be performed more smoothly. Accordingly, an underwater handling tool was developed that can be inserted while offsetting the tension of the spring. Using this handling tool applies a load to the dummy rod and rotates the dummy rod itself, offsetting the tension of the spring and allowing the dummy rod to be inserted without bending. This handling tool is equipped with a shock absorbing device to protect the dummy rod and spring, and a module to rotate the dummy rod. As a result of inserting the dummy rod using the developed handling tool, it was possible to easily insert the dummy rod into unit cells that were previously impossible to insert.

Keywords: Dummy rod, Rod insertion, Fuel assembly, Unit cell

Evaluation of Compression Characteristics of Guide Tube for Fuel Assembly Restoration

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Nuclear fuel assemblies are exposed to high temperature and high pressure environments underwater for long periods of time in a reactor, leading to deterioration of the assembly structure. These assembly consists of fuel rods, grids, a top nozzle, a bottom nozzle and guide tubes. In particular, the integrity of the guide tube made of Zircaloy-4 is a very important part in handling the assembly. In the Post Irradiation Examination Facility (PIEF), there are 14×14 Westinghouse STD assemblies that have lost their handleability due to the top nozzle being removed for damaged fuel rod test. To handle these assemblies, it is reasonable to use cut guide tubes whenever possible. Therefore, it is necessary to determine the irradiation embrittlement state of the guide tube before designing or manufacturing parts that can connect the top nozzle and the guide tubes. Therefore, in this paper, the location for installing the top nozzle-guide tube connection parts was selected in the height range of 3,460 to 3,713 mm, and guide tube specimens were made within that range. Offset strain was derived from the load-displacement curve obtained through compression testing to confirm whether the ductility of guide tubes was maintained. As a result, there was no significant difference in strength and ductility of the guide tube within the above length range. In addition, it was confirmed that the ductility was maintained enough to install the top nozzle-guide tube connection parts. Therefore, it is judged that there will be no problem even if the top nozzle-guide tube connection parts are installed in the guide tube to restore the handleability of the assemblies.

Keywords: Guide tube, Restoration handleability, Fuel assembly, Anchor bolt

Fracture Evaluation of Spent Nuclear Fuel Storage Canister

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In this study, a fracture evaluation of the spent nuclear fuel storage canister was conducted. Stainless steel alloys are typically used as the material for canisters, and therefore, a separate destructive evaluation is not required for safety analysis reports. However, in this research, a methodology for conducting a destructive evaluation was proposed for assessing the acceptability of cracks detected during in-service inspections for long-term storage due to reasons such as stress corrosion cracking.

For the fracture evaluation, analytical equations provided in the design code such ASME were employed, and finite element method (FEM) based linear elastic fracture mechanics (LEFM) was performed to validate the effectiveness of the analytical equations. Impact analyses such as tip-over of the storage cask on a concrete pad were performed, and the fracture evaluation using stresses resulting from the impact analysis under accident conditions and residual stresses from welds were carried out. Through this research, geometric dimensions for cracks exceeding the fracture criteria were established.

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Keywords: Spent nuclear fuel, Storage of spent nuclear fuel, Canister, Fracture evaluation

Comparison and Validation Between Analysis and Test for Vibration Evaluation of Nuclear Fuel Assembly

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A comparison and validation between the analysis and vibration test data of a nuclear fuel assembly were conducted. During the comparison and validation process, various parameters that govern the vibration behavior of the fuel assembly were determined, including nuclear fuel rod's stiffness, spring constants of the dimple and spring of support structures, and damping coefficients.

The calibration of the vibration analysis model aimed to find analysis parameters that can accurately simulate the vibration behavior of the test data. For calibration, power spectral density (PSD) diagrams were generated for both the measured signals from the test and the calculated signals from the analysis. The correlation coefficient between these two PSD plots was calculated. To find the analysis parameters, each parameter was defined as a variable with an appropriate range. Latin hypercube sampling was used to generate multiple sample points in the variable space. Analysis was performed for the generated sample points, and PSD plot correlation coefficients were calculated. Using the generated sample points and their corresponding results, a Gaussian Process Regression model was implemented for PSD plot correlation coefficients and the maximum PSD value. Based on the constructed surrogate model, the optimal analysis parameters were easily found without additional computations.

Through this method, it was confirmed that the analysis model using the optimal parameters appropriately simulates the vibration behavior of the test.

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Keywords: Nuclear fuel assembly, Vibration, Vibration test, Comparison, Vibration analysis

Develop a Simplified Model for Cask-Level Drop Analysis and Calculate Fuel Rod Failure Probability

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For safe and economical spent fuel management, assessing the integrity of the cladding, which is the first barrier to the escape of radioactive material, is very important. For the sake of risk assessment, it is essential to calculate the probability of failure of the spent fuel rods loaded inside the cask during the transportation or storage. However, due to the large amounts of calculations required, it is not practical to analyze every detail of the spent fuel rods and assemblies. This study presents a methodology to perform a cask-level analysis by sequentially simplifying the fuel rods and spent fuel assemblies for the calculation of fuel rod failure probability. A simplified single fuel rod model was generated by considering the material properties of a high burnup fuel rod stored in dry storage for approximately 5 years and the interfacial bonding conditions of the cladding tube. The simplified model produces the same deflection as the detailed model at the critical moment that produces a fracture plastic strain of 1%. The developed single fuel rod simplified model is assembled in a CE 16×16 configuration, and a methodology is presented in which the CE 16×16 assembly model is once again replaced by a simplified model with a cuboidal shape. Compression analyses were performed on each part of the CE 16×16 model to obtain isotropic property data, and a simplified model was created based on those data and the cross-sectional second moment values of the parts. A cask drop analysis was performed to validate the similarity of the CE 16×16 model and the simplified model by comparing important structural responses such as impact acceleration. The 20 simplified fuel assembly models and one detailed model were loaded into a cask to perform the drop analysis. For the detailed model, the impact acceleration was extracted for different loading positions and the corresponding impact load and pinch load were derived. The spring force and contact force corresponding to the pinch load were extracted by applying a Python script technique to extract the maximum value of them exerted on each fuel rod. The vulnerability of spent fuel rods to bending loads and the failure criteria were considered during the simplification process of a single fuel rod. From the extracted impact and pinch loads, the probability of failure of the spent fuel rods as a function of impact acceleration can be calculated.

Acknowledgements

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Keywords: Risk assessment, Simplified model, Impact acceleration, Fuel failure probability

Reactive-Crystallization Method for Purification of LiCl-KCl Eutectic Salt Waste

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The nuclide management process for reducing the environmental burden being developed by the Korea Atomic Energy Research Institute is performed in molten salts, resulting in contaminated salt wastes containing fission products such as Cs, Sr, Ba, and rare-earth nuclides. In addition, the spent fuel of a molten salt reactor (MSR) contains a variety of fission products, and a purification process may be required for the reuse of the salt and the separation and disposal of the fission products in the spent nuclear fuel. The melt-crystallization method is a technique used for the purification and separation of chemicals or metals based on the different melting points of the different substances. In a recent study, our group developed a reactive-crystallization method using Li_2CO_3 precipitation agent to precipitate metal corrosion from the reactor through a chlorination reaction by HCl and Cl_2 , which may occur in chloride molten salt, and successfully precipitated the metal precipitate and purified and recovered LiCl salt. In this study, reactive-crystallization method has been established for removing fission products and corrosive materials. Using the reactive crystallization method, white LiCl-KCl salt that was not discolored by metal corrosion was recovered through the crystallization plates, and fission products and metal elements were shown to be suppressed to several ppm in the purified salt. Consequently, high-purity salts were recovered with high nuclide and corrosive separation efficiencies. The reactive crystallization procedure can also be applied to other salt waste systems, such as MSR nuclear fuel treatment and molten salt chemistry for the elimination of corrosive substances.

Keywords: LiCl-KCl, Eutectic salt, Salt purification, Reactive-crystallization

Development of an Integrated Nuclear Fuel (GIFT) and Thermal Simulation (COBRA-SFS) Code for Dry Storage Analysis and Its Application to Safety Assessments

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Dry storage of nuclear fuel is compromised by threats to the cladding integrity, such as creep and hydride reorientation. To predict these phenomena, spent fuel simulation codes have been developed. In spent fuel simulation, temperature information is the most influential factor for creep and hydride formation. Traditional fuel simulation codes required a user-defined temperature history input which is given by separate thermal analysis. Moreover, geometric changes in nuclear fuel, such as creep, can alter the cask's internal subchannels, thereby changing the thermal analysis. This necessitates the development of a coupled thermal and nuclear fuel analysis code.

In this study, we integrated the 2D FDM nuclear fuel code GIFT developed at SNU with COBRA-SFS. Using this, we analyzed spent nuclear stored in TN-24P dry storage cask over several decades and identified conditions posing threats due to phenomena like creep and hydrogen reorientation, represented by the burnup and peak cladding temperature at the start of dry storage. We also investigated the safety zone of spent nuclear fuel based on burnup and wet storage duration using decay heat.

Keywords: Dry storage, Spent fuel simulation, COBRA-SFS, Code coupling, Cladding integrity

Characterization of Reaction-Prevent Materials for the Reduction of Radioactive Waste From Metallic Nuclear Fuel Casting

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Currently, the most promising fuel candidate for use in sodium fast reactors (SFRs) is metallic fuel, which is produced by a modified casting method in which the metallic fuel material is sequentially melted in an inert atmosphere to prevent volatilization, followed by melting in a graphite crucible, and then injection casting in a quartz (SiO_2) mold to produce metallic fuel slugs.

In previous studies, U-Zr metallic fuel slugs have been cast using Y_2O_3 reaction prevent coatings. However, U-Zr alloy-based metallic fuel slugs containing highly reactive rare earth (RE) elements are highly reactive with Y_2O_3 -coated quartz (SiO_2) molds and form a significant thickness of surface reaction layer on the surface of the metallic fuel slug. Cast parts that have reacted with nuclear fuel materials become radioactive waste.

To decrease amount of radioactive waste, advanced reaction prevent material was developed. Each RE (Nd, Ce, Ln, Pr) element was placed on the reaction prevent material and thermal cycling experiments were carried out.

In casting experiments with U-10wt% Zr, it was reported that Y_2O_3 layer has a high reaction prevent performance. Therefore, the reaction layer properties for RE elements with higher reactivity than uranium elements were evaluated. To investigate the reaction layer between RE and NdYO_3 , the reaction composition and phase properties as a function of RE content and location were investigated using SEM, EDS, and XRD. The results showed that NdYO_3 ceramics had better anti-reaction performance than Y_2O_3 .

Keywords: Metallic fuel slug, Injection casting, High temperature material

Development of Graphite-Boron Composites With Enhanced Thermal Conductivity and Radiation Shielding Property for Dry Spent Nuclear Fuel Storage Application

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The objective of this study is development of graphite-boron composite material as a replacement for metal canisters to Improve the heat dissipation and radiation shielding performance of dry spent nuclear fuel storage system and reduce the volume of waste storage system. KEARI research team plan to use the graphite matrix manufacturing technology to pelletize the graphite matrix and adjust the content of phenolic resin binder to minimize pore formation. Specifically, we plan to adjust the ratio of natural and synthetic graphite powder and use uniaxial pressing technology to manufacture black graphite matrix with extremely high radial thermal conductivity. After optimizing the thermal conductivity of the graphite matrix, we plan to mix it with selected boron compounds, shape it, and perform sintering and purification heat treatments at high temperatures to manufacture standard composite materials.

Keywords: Spent nuclear fuel, Dry cask storage, Graphite-boron composite, Metallic canister

Uncertainty Quantification for the Thermal Analysis of the CANDU Spent Fuel Dry Storage Silo

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In our previous study, we developed a CFD thermal analysis model for a CANDU spent fuel dry storage silo. The purpose of this model is to reasonably predict the thermal behavior within the silo, particularly Peak Cladding Temperature (PCT), from a safety perspective. The model was developed via two steps, considering optimal thermal analysis and computational efficiency. In the first step, we simplified the complex geometry of the storage basket, which stored 2,220 fuel rods, by replacing it with an equivalent heat conductor with effective thermal conductivity. Detailed CFD analysis results were utilized during this step. In the second step, we derived a thermal analysis model that realistically considered the design and heat transfer mechanisms within the silo.

We developed an uncertainty quantification method rooted in the widely adopted Best Estimate Plus Uncertainty (BEPU) method in the nuclear industry. The primary objective of this method is to derive the 95/95 tolerance limits of uncertainty for critical analysis outcomes. We initiated by assessing the uncertainty associated with the CFD input mesh and the physical model applied in thermal analysis. And then, we identified key parameters related to the heat transfer mechanism in the silo, such as thermal conductivity, surface emissivity, viscosity, etc., and determined their mean values and Probability Density Functions (PDFs). Using these derived parameters, we generated CFD inputs for uncertainty quantification, following the principles of the 3rd order Wilks' formula. By calculating inputs, A database could be constructed based on the results. And this comprehensive database allowed us not only to quantify uncertainty, but also to evaluate the most conservative estimates and assess the influence of parameters.

Through the aforementioned method, we quantified the uncertainty and evaluated the most conservative estimates for both PCT and MCT. Additionally, we conducted a quantitative evaluation of parameter influences on both. The entire process from input generation to data analysis took a relatively short period of time, approximately 5 days, which shows that the developed method is efficient.

In conclusion, our developed method is effective and efficient tool for quantifying uncertainty and gaining insights into the behavior of silo temperatures under various conditions.

Keywords: CANDU spent fuel, Dry storage silo, Thermal analysis, Uncertainty quantification, BEPU

Analysis and Design of Mechanical Decladding for Spent Fuel in Head-End

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The PRIDE scale mechanical decladder is decladding apparatus for separating and recovering fuel material and cladding hull by horizontally slitting rod-cut.

In order to enhance mechanical decladding efficiency, the main requirements were considered as follows. Decladding of the fuel rods may be performed by rotation of three circular cutting blades inserted among the rollers arranged at 120° portion. In a mechanical decladder, a slitting assembly as a unit for slitting the cladding tube may include cutting blades for slitting and rollers for guiding extrusion of the cladding tube. Rotation of the cutting blades may be caused by the fuel rods being extruded from a plurality of rollers. Slitting intervals of rod-cuts having different diameters may be controlled by adding or removing a spacing plate between the cutting blade and a ranch bolt for fixing the slitting blade to the slitting assembly. An extrusion velocity with respect to the fuel rods may be controlled by a hydraulic pressure applied to the fuel rods. A force for cutting the fuel rods may be adjusted by controlling steel plates. Forces applied to a plurality of rollers may be generated by the hydraulic cylinder. The hydraulic pressure may be controlled by hydraulic pressure controller.

The PRIDE scale mechanical decladder mainly consists of auto feeding module, hydraulic cylinder module and blade module. A load cell was installed between the hydraulic cylinder and the extrusion pin to measure the decladding force and slitting velocity, and a data acquisition system capable of obtaining data by using the RSC 232 was constructed. Also, the control panel can control the forward and backward movement of the extrusion pin, the hydraulic flow rate, and the hydraulic velocity.

In the mechanical decladding test, 40 pieces of simulated rod-cuts were loaded in two auto feeding basket and slit by utilizing the 3-CUT blade modules in the housing, and hulls and simulated pellets were collected in the collection container. As a result, 80 pieces of simulated rod-cut (brass pellets + Zry4 tube) were slit continuously without any problem. About 35 min was required to slit 80 rod-cuts and average decladding force was 260 kg. The decladding force of the ceramic simulated rod-cuts (castable) requires 25 kg less force than the brass pellets. Therefore, it is estimated that the spent fuel rod-cut can be fully split into three pieces using the mechanical decladder.

Acknowledgements

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Keywords: Design, Mechanical, Decladding, Spent, Fuel, Rod

Analysis of High-Enriched Core Composition Behavior in Micro Modular Reactor During Long-Cycle Operation

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Recently, as carbon-neutral energy sources become increasingly important worldwide, SMRs (Small Modular Reactors), which offer significantly enhanced safety, versatility, and mobility compared to conventional nuclear reactors, are gaining attention as a viable alternative. SMR generally refers to small modular reactors with a power output of 300 MWe or less. Unlike conventional reactors, SMRs are characterized by an all-in-one design where peripheral systems and equipment are all integrated into the reactor itself, leading to enhanced reliability and durability. Additionally, the nuclear fuel reloading cycle is significantly extended compared to traditional reactors, resulting in a substantial reduction in maintenance difficulty and costs. Researchers have taken note of these characteristics of SMRs, particularly the extended fuel reloading cycle.

Therefore, we have initiated the initial design of an ultra-small Micro Modular Reactor with an electricity generation capacity of 10 MWe and a fuel cycle of up to 55 years, with the goal of using it as a propulsion power source for various transportation modes, especially ships. Our design of MMR, called 'ARA,' is primarily distinguished by its use of U233 and Th232 fuels instead of conventional UO₂ fuel.

Due to various features of 'ARA,' including different fuel compositions, ARA is predicted to exhibit several characteristic features compared to conventional PWRs. In this study, among these characteristics, we focused on predicting changes in material composition within the fuel rod during the extended cycle operation of high-enriched fuel, rather than short-cycle operation using low-enriched fuel, unlike conventional reactors.

The primary goal of this research is to observe the behavior of the composition of the materials used in the fuel cycle of the MMR, which utilizes U233 and Th232 fuels instead of UO₂. Considering the difficulties in the spent nuclear fuel disposal process, many different trials were made to minimize the fission products of ARA, which differs from conventional reactors in terms of fuel type, size, and fuel cycle, in relation to waste generation.

Keywords: Micro modular reactor, Small modular reactor, Long-life nuclear fuel, Nuclear transmutation

Fabrication of SimFuel Containing CsI Based on High-Pressurizing Sintering

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In KAERI, the nuclide management technology is currently being developed for the reduction of disposal area required for spent fuel management. Among the all fission products of interest, Cs, I, Kr, Tc are considered to be significantly removed by following mid-temperature and high-temperature treatment, however, a difficulty of real spent-fuel thermal treatment experiment limits the development of such thermal treatment. The test employing SimFuel (Simulated Spent Fuel) can be an alternative for such condition, however, the fabrication of SimFuel containing semi-volatile species such as Cs, I and Re (substitute for Tc) was not achieved for conventional sintering method since such species are easily removed during hot temperature treatment. In this study, for the prevention of volatilization of such species and the inclusion of semi-volatile species in fabrication of SimFuel, argon-based high pressurizing up to Max 100 bar was considered to be applied in high temperature treatment. For this, lab-scale hot-isostatic press applicable up to 1,500°C was fabricated and is being waiting for the approval for high-pressure test. After approval of license, UO_2 based SimFuel containing CsI will be fabricated and its micro-structure and composition will be evaluated through SEM-EDX and XRD

Acknowledgements

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Keywords: SimFuel, CsI, Gas pressurizing sintering, Semivolatile

Design and Implementation of a Thermal Analysis Framework for Solidified Waste Form Containing Heat-Generating Nuclides

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The design and fabrication of suitable waste forms with high thermal and structural stability are essential for the safe management and disposal of radioactive wastes. In particular, the thermal properties and temperature distribution of waste form containing high heat-generating nuclides such as Cs and Sr can be used to evaluate its thermal stability, but also provide useful information for the design of canisters, storage systems, and repositories.

In this study, a new program code-based thermal analysis framework has been developed to facilitate the characterization, design, and optimization of the waste form. Matlab was used as a software development platform because it provides powerful mathematical computation and visualization components such as the partial differential equation (PDE) toolbox for solving heat transfer problems using finite element method, the App Designer for graphical user interface (GUI), and the MATLAB Compiler for sharing MATLAB programs as standalone applications and web applications. The thermal analysis results such as temperature distribution, heat flux, maximum/minimum temperature, and centerline/surface temperature profile are visualized with graphs and tables. To evaluate the effectiveness of the developed program, several design and optimization studies were carried out for the SrTiO₃ waste form, selected as a stable form of strontium nuclide.

Acknowledgements

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Keywords: Thermal analysis framework, Cylindrical waste form, Heat-generating nuclide, MATLAB app designer

Maintenance of Hydrogen Analyzer for Zr Cladding of Spent Fuel

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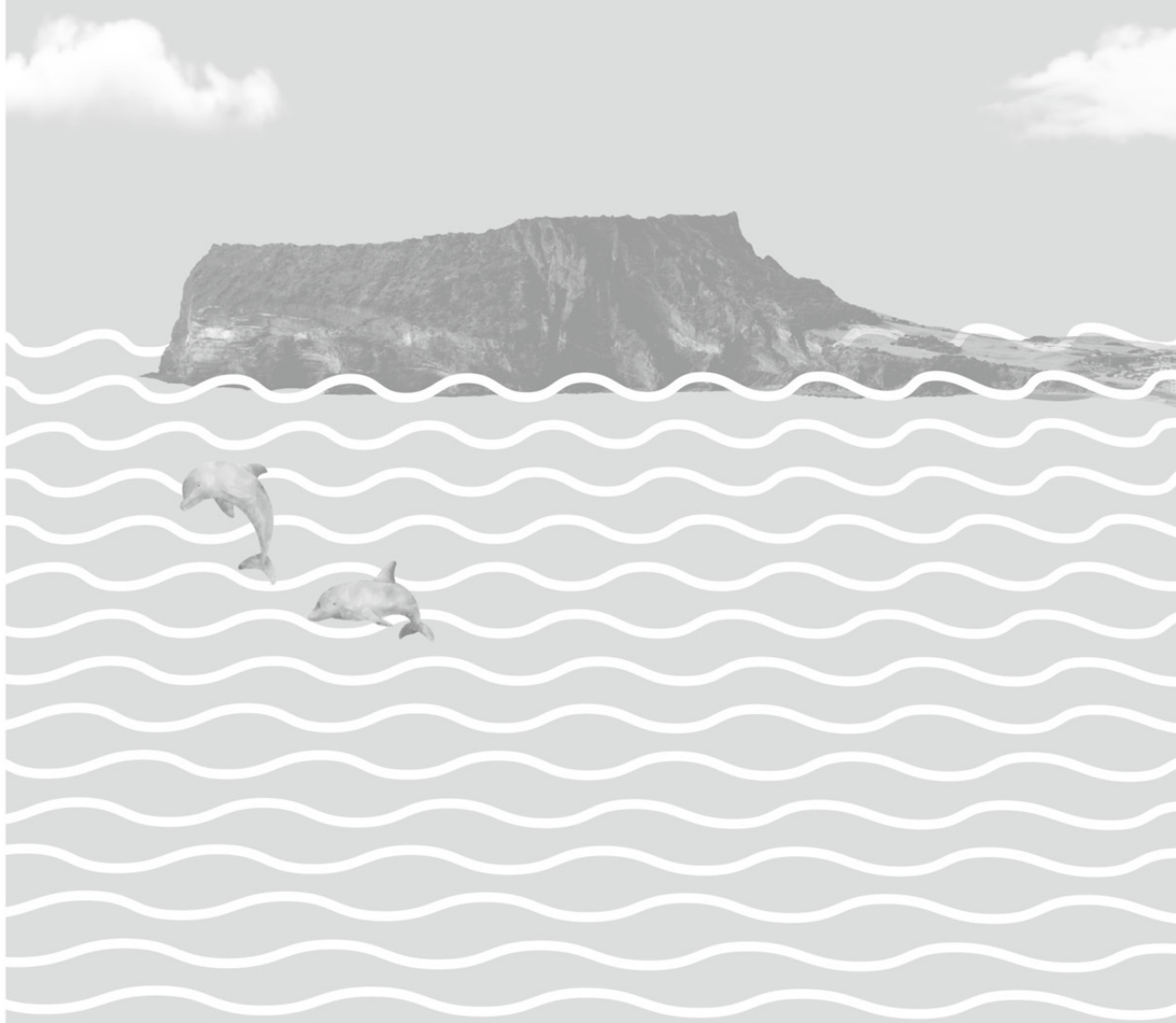
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Korea Atomic Energy Research Institute (KAERI) has been operating the Post Irradiation Examination Facility (PIEF). The facility has many PIE equipment and one of them is a hydrogen analyzer for measuring hydrogen contents in Zr cladding of spent fuel. The cladding tube of fuel is oxidized in the core environment of high temperature and pressure and absorbs some of the hydrogen generated during the oxidation. The hydrogen content increases with the increase of burn-up, and causes hydriding of the material, which degrades the mechanical properties. Therefore, hydrogen content analysis of the cladding tube is required for the performance and integrity evaluation of spent fuel. In PIEF, the hydrogen analyzer extracts hydrogen gas from Zr cladding by the hot extraction method. The hydrogen gas flows with inert gas and oxidizes to H₂O through a CuO reagent. Finally, an IR detector measures the hydrogen amount from the absorbed IR intensity at a specific wavelength. Because the equipment is in the glove box and has some consumable parts, the maintenance work was performed as a radiation work.

Keywords: Spent fuel, Zr cladding, Hydrogen analysis, Oxidation, Infrared absorbance

3분과

고준위폐기물처분 (ORAL)



Comparative Analysis of Terminology Definitions for Spent Nuclear Fuel Disposal in Domestic Regulations With Related Foreign Definitions

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As part of the preparation of a glossary of terminologies related to the disposal of spent nuclear fuel, definitions of potentially issuable terminologies used in domestic regulations were inferred from relevant regulations or comparatively analyzed with foreign definitions.

These terminologies are safety assessment and performance assessment, safety function and safety performance, disposal containers and package, isolation and containment, and so on.

Their concise and easy-to-understand definitions have been proposed in order to obtain these opinions of stakeholders.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) grant funded by the Korea government (Compilation of glossary related to storage and disposal of spent nuclear fuel).

Keywords: Assessment, Isolation, Safety, Performance, Terminology

Review the Corrosion Mechanism in the Spent Nuclear Fuel Cask Through the Mass-Balance Equations and Chemical Reactions

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For the sake of future generations, the management of radioactive waste is essential. The disposal of spent nuclear fuel (SNF) is considered an urgent challenge to ensure human safety by storing it until its radioactivity drops to a negligible level. Evaluating the safety of disposal facilities is crucial to guarantee their durability for more than 100,000 years, a period sufficient for SNF radioactivity to become ignored. Past studies have proposed various parameters for forecasting the safety of SNF disposal. Among these, radiochemistry and electrochemistry play pivotal roles in predicting the corrosion-related chemical reactions occurring within the SNF and the structural materials of disposal facilities.

Our study considers an extreme scenario where the SNF canister becomes compromised, allowing underground water to infiltrate and contact the SNF. We aim to improve the corrosion mechanism and mass-balance equation compared with what Shoesmith et al. proved under the same circumstances. To enhance the comprehensibility of the chemical reactions occurring within the breached SNF canister, we have organized these reactions into eight categories: mass diffusion, alpha radiolysis, adsorption, hydrate formation, solidification, decomposition, ionization, and oxidation. After categorization, we define how each species interacts with others and calculate the rate of change in species' concentrations resulting from these reactions.

By summing up the concentration change rates of each species due to these reactions, we redefine the mass-balance equations for each species. These newly categorized equations, which have not been explained in detail previously, offer a detailed description of corrosion reactions. This comprehensive understanding allows us to evaluate the safety implications of a compromised SNF canister and the associated disposal facilities by numerically solving the mass-balance equations.

Acknowledgements

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Keywords: Spent nuclear fuel, Corrosion mechanism, Chemical reaction, Mass-balance equation

Case Study of Deep Borehole Drilling by Rock Types for Deep Geological Disposal of HLW

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Deep borehole drilling is essential not only to select the host rock type for deep geological disposal of high-level radioactive waste (HLW), but also to identify the characteristics of the disposal site during the site selection process. In particular, since the disposal depth of HLW is considered to be over 300 m, deep borehole drilling must be performed. In deep borehole drilling, drilling design, excavation, and operation may vary depending on the rock type, drilling depth, and drilling purpose etc. This study introduced cases in which Korea was divided into four geotectonic structures and four representative rock types and conducted with a goal of 750 m drilling depth. Prior to this, a review of deep drilling cases conducted at domestic and abroad was presented. If sufficient time and cost are available, several drilling holes can be excavated for various purposes, but if not, one or two drilling holes should be used to achieve the objectives of various fields related to HLW disposal. The presence of bedding, strata or fault zones depending on the type of rock, etc. may affect drilling deviation or circulating water management. In addition, unlike drilling in general geotechnical investigation drilling, the use of polymers or grouting agents is limited to determine hydraulic and geochemical characteristics. This report introduces the experience considered during the design and drilling process of deep drilling in granite, gneiss, sedimentary rock, volcanic rock, etc., and is expected to be used as basic data when carrying out future HLW projects.

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Keywords: Deep borehole drilling, High-level radioactive waste, Rock type, Drilling water management

Preliminary Concept of Operation Process of Encapsulation Plant in Deep Geological Repository

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Spent nuclear fuels (SNFs) are stored in nuclear power plants for a certain period of time and then transported to an interim storage facility. After that, SNFs are finally repackaged in a disposal canister at an encapsulation plant for final disposal. Finland and Sweden have already completed the design of the spent nuclear fuel encapsulation plant. In particular, Finland has begun the construction of the encapsulation plant and is on the verge of completion. Korea Radioactive Waste Agency (KORAD) is conducting a conceptual design of a deep geological repository for SNFs. Conceptual design of the encapsulation plant is part of the research activity. It is highly required to draft an operation process of the encapsulation plant before an actual design activity. As part of the activity, Finnish design concept of the encapsulation plant and experience were thoroughly reviewed. Finally a preliminary concept of the operation process was proposed considering Korean unique situations such as the volume of SNFs estimated to be disposed of, types of transportation cask and other considerations.

Keywords: Spent nuclear fuel, Encapsulation, Disposal, Transportation cask, Operation process

Hydrogeochemical Factors Related to Uranium Concentration Levels in Granitic Aquifers in Korea

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Uranium (U) is a hazardous material that can lead to both chemical and radiological toxicity, including kidney damage and health issues associated with radiation exposure. In South Korea, where shallow weathered granitic aquifers are widespread, several previous studies have reported high levels of radioactivity in shallow groundwater. This ultimately led to the closure of 60 out of 4,140 groundwater production wells in South Korea. In this study, we examined aquifers currently dedicated to drinking water supply and investigated a dataset of 11,225 records encompassing 103 environmental parameters, based on the random forest classifier. This dataset comprises 80 physical parameters associated with the hydraulic system and 23 chemical parameters linked to water-rock interactions. Among the hydraulic parameters, the presence of a coarse loamy texture in the subsoil displayed a notable positive relationship with the concentration of uranium, implying that it plays a significant role in forming redox conditions for the leaching of uranium from host rocks. Fluorine (F), a major product of water-rock interaction in granitic aquifers, exhibited a positive correlation with the distribution of uranium concentrations. The positive relationship between F concentration and uranium levels suggests that the dissolved uranium originates from groundwater interacting with granites. In conclusion, our findings indicate that two key factors, namely the infiltration capacity of soil layers and the aqueous speciation in groundwater resulting from interactions with local solids, play important roles in determining uranium concentrations in granitic aquifers.

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Keywords: Uranium, Machine learning, Random forest, Groundwater, Emerging contaminants

Development of HADES Code for Assessing the Performance of Disposal Repositories for Spent Nuclear Fuel

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The HADES (High-level rAdiowaste Disposal Evaluation Simulator) was developed by the Nuclear Fuel Cycle & Nonproliferation (NFC) laboratory at Seoul National University (SNU), based on the MOOSE Framework developed by the Idaho National Laboratory (INL). As an application of the MOOSE Framework, the HADES incorporates not only basic MOOSE functions, such as multi-physics analysis using Finite Element Method (FEM) and various solvers, but also additional functions for estimating the performance assessment of Deep Geological Repositories (DGR). However, since the MOOSE Framework does not have complex mesh generation and data analyzing capabilities, the HADES has been developed to incorporate these missing functions. In this study, although the Gmsh, finite element mesh generation software, and Paraview, finite element analysis software, were used, other applications can be utilized as well.

The objectives of HADES are as follows: (i) assessment of the performance of a Spent Nuclear Fuel (SNF) disposal system concerning Thermal-Hydraulic-Mechanical-Chemical (THMC) aspects; (ii) Evaluation of the integrity of the Engineered Barrier System (EBS) of both general and high-efficiency design perspective; (iii) Collaboration with other researchers to evaluate the disposal system using an open-source approach. To achieve these objectives, performance assessments of the various disposal systems and BMTs (BenchMark Test), conducted as part of the DECOVALEX projects, were studied regarding TH behavior. Additionally, integrity assessments of various DGR systems based on thermal criteria were carried out. According to the results, HADES showed very reasonable results, such as evolutions and distributions of temperature and degree of saturation, when compared to validated code such as TOUGH-FLAC, ROCMAS, and OGS (OpenGeoSys). The calculated data are within the range of estimated results from existed code. Furthermore, the first version of the code, which can estimate the TH behavior, has been prepared to share the contents using Git software, a free and open-source distribution system.

Acknowledgements

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Keywords: HADES, MOOSE framework, Performance assessment, Open-source

Change of Electrical Resistance According to Saturation and Erosion of Bentonite Buffer: Numerical Study

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The concept of deep geological disposal for high-level radioactive waste is based on an engineered barrier system (EBS), including a canister, bentonite buffer and backfill material. The bentonite buffer is key component of the EBS to prevent groundwater infiltration and radionuclide leakage. However, the bentonite buffer can become saturated due to groundwater flow through the excavation damaged zone in the adjacent rock, causing erosion of bentonite buffer and affecting the long-term performance of EBS. While the RH (relative humidity) sensor is commonly used to assess the degree of saturation in the bentonite buffer, it has a critical challenge due to its sensor size, which can disturb the overall integrity of the bentonite buffer during the initial installation process. In contrasts, the electrical resistivity test, widely known as a non-destructive method, is used to predict soil properties such as the degree of saturation and water contents. This method measures the electric resistance of materials using electric current induced by electric potential difference between two electrodes. Notably, there is no study that assess the integrity of bentonite buffer in a nuclear waste repository using electrical resistivity measurement.

This study presents the electrical resistance numerical module under steady state using commercial finite element method (FEM), and quantitatively estimate the change of electrical resistance according to saturation and erosion of bentonite buffer. Furthermore, the electric potential and current density distribution formed between two electrodes are analyzed.

Acknowledgements

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Keywords: Engineered barrier system, Bentonite buffer, Erosion, Electrical resistivity test, Electrical resistance numerical module

Characterizing Unsaturated Zone for an Infiltration Test in the UNIT Site Near KAERI Underground Research Tunnel

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Long-term climate and surface environment changes can influence the geological subsurface environment evolution. In this context, a fluid flow pathway developing and connection possibility can be increased between the near-surface zone and deep depth underground. Thus, it is necessary to identify and prepare for the overall fluid flow at the entire geological system to minimize uncertainty on the spent nuclear fuel (SNF) disposal safety.

The fluid flow outside the subsurface environment is initially penetrated through the surface and then the unsaturated area. Thus, the previously proved reports, POSIVA in Finland, suggested that sequential research about the fluid infiltration experiment (INEX) and the investigation is necessary. Characterizing the unsaturated zone can help predict changes and ensure the safety of SNFs according to geological long-term evolution. For example, the INEX test was conducted at the upper part of ONKALO, about 50 to 100 m depth, to understand the geochemical evolution of the groundwater through the unsaturated zone, to evaluate the main flow of groundwater that can approach the SNF disposal reservoir, and to estimate the decreasing progress of the buffering capacity along the pathway through the deep geological disposal.

In the present study, a preliminary test was performed in the UNsaturated-zone In-situ Test (UNIT) facility near the KAERI underground research tunnel to design and establish a methodology for infiltration experiments consistent with the regional characteristics. The results represented the methodological application is possible for characterizing unsaturated-zone to perform infiltration experiments. The scale of the experiment will be expanded sequentially, and continuous research will be conducted for the next application.

Acknowledgements

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Keywords: Characterization, Unsaturated zone, Infiltration, UNIT site, Preliminary test

Analysis of Natural Rocks Collected From Uranium Deposits in Okcheon Metamorphic Zone Using Raman Spectroscopy

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Raman characteristics of various minerals constituting natural rocks collected from uranium deposits in Okcheon metamorphic zone in Korea are presented. Micro-Raman spectra were measured using a confocal Raman microscope (Renishaw in Via Basis). The focal length of the spectrometer was 250 mm, and a 1800 lines/mm grating was installed. The outlet of the spectrometer was equipped with a CCD (1,024×256 pixel) operating at -70°C. Three objective lenses were installed, and each magnification was ×10, ×50, and ×100 times. The diameter of the laser beam passing through the objective lens and incident on the sample surface was approximately 2 μm. The laser beam power at 532 nm was 1.6 mW on the sample surface. Raman signal scattered backward from the sample surface was transmitted to the spectrometer through the same objective lens. To accurately determine the Raman peak position of the sample, a Raman peak at 520.5 cm⁻¹ measured on a silicon wafer was used as a reference position.

Since quartz, calcite, and muscovite minerals are widely distributed throughout the rock, it is easy to observe with an optical microscope, so there is no difficulty in measuring the Raman spectrum. However, it is difficult to identify the uraninite scattered in micrometer sizes only with a Raman microscope. In this case, the location of uraninite was first confirmed using SEM-EDS, and then the sample was transferred to the Raman microscope to measure the Raman spectrum. In particular, a qualitative analysis of the oxidation and lattice conditions of natural uraninite was attempted by comparing the Raman properties of a micrometer-sized natural uraninite and a laboratory-synthesized UO₂ pellet. Significantly different T_{2g}/2LO Raman intensity ratio was observed in the two samples, which indicates that there are defects in the lattice structure of natural uraninite. In addition, no uranyl mineral phases were observed due to the deterioration of natural uraninite. This result suggests that the uranium deposit is maintained in a reduced state. Rutile is also scattered in micrometer-sizes, similar to uraninite. The Raman spectrum of rutile is similar in shape to that of uraninite, making them confused. The Raman spectral differences between these two minerals were compared in detail.

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Keywords: Raman spectroscopy, SEM-EDS, Natural uraninite, UO₂, Uranium deposits

Integration of FEPs for Development of Scenarios

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According to the analysis of the Korean Radioactive Waste Society, saturation of nuclear power plant temporary storage is expected sequentially from 2031, and accordingly, the need for high-level radioactive waste disposal facilities has emerged.

In order 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. For safety analysis, it needs a collection of arguments and evidence. and IAEA defined it as ‘Safety case’.

The Systematic method, which derives scenarios by systematizing and combining possible phenomena around the repository, is widely used for developing Safety case. Systematic methods make use of the concept of Features, Events and Processes (FEP). FEP identifies features that affect repository performance, events that can affect a short period of time, and processes that can have an impact over a long period of time.

Many countries, such as Finland, Sweden, Japan, United States, etc., are in process of licensing disposal facilities by using ‘Safety case’. And they then develop their own project-specified FEP lists and employ them for performing safety assessments. However, the systematic procedure for generating scenarios for safety evaluation is not clearly defined.

According to the International Atomic Energy Agency (IAEA) Safety Standards Series (SSG-23), the bottom-up method is an approach for conducting safety analysis using Features, Events, and Processes (FEPs). However, the process of how each FEP is utilized to establish a scenario for safety evaluation remains unclear. Additionally, there exists not only a bottom-up approach for generating scenarios using FEPs, but also a hybrid scenario development method that incorporates a top-down approach based on safety functions. Each country address scenario derivation in accordance with the adopted hybrid method. Nevertheless, a challenge arises in its application due to discrepancies between their approach and the hybrid approach specific which we are going on.

Hence, this study introduces the FEP integration methodology for generating scenarios based on the hybrid scenario development method using the FEP list.

Keywords: High-level radioactive waste, Disposal, Safety case, FEP, Bottom-up

Solute Mixing in Rough-Walled Fracture Intersections at High Peclet Number

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The safe disposal of high-level radioactive waste has become a prominent global concern, necessitating rigorous safety assessments for deep geological disposal facilities. In Korea, crystalline rock with low-permeability is considered as the host rock for radioactive waste disposal, and fluid flow and solute transport in a low-permeability rock formation predominantly occur through interconnected fracture network. To analyze and predict fluid flow and solute transport behavior within the fractures, a comprehensive understanding of solute mixing at fracture intersections is crucial. However, difficulty in direct observation of the mixing processes occurring within microscale fracture intersections has led only to analytical and numerical studies, which requires thorough experimental study based on direct observations and measurements for a fundamental understanding of the mixing processes in fracture intersections.

In this study, elaborate experiments are being prepared and conducted to measure the complex flow velocity/structure and solute concentration at rough-walled fracture intersections, using a microscale visualization technique of micro Particle Image Velocimetry (micro-PIV) system. Most analytical and numerical studies have shown that at high Peclet number (Pe) $> 1,000$, streamline-routing model plays a major role in redistributing solutes at the fracture intersection, at which the mixing ratio converges to zero. As opposed to the conventional mixing model, our experiments found the rebounding of the mixing ratio in the inertial flow regime, indicating an enhanced solute mixing at the intersection. Flow visualization has demonstrated that the inertial flow features, such as the development of large-scale eddies and the straightening of main streamlines, enhance the physical mixing of solutes at rough-walled fracture intersections. The findings provide insights into the influence of fracture geometry on flow dynamics and its significant impact on solute mixing at fracture intersections.

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Keywords: Fracture, Fracture intersection, Solute mixing, Microscopic particle image velocimetry (micro-PIV), Flow visualization

Hydraulic Properties of the WRK Compacted Bentonite Core as an Engineering Barrier Material in the SNF Repository

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The WRK (Waste Repository Korea bentonite) compacted bentonite medium has been considered as the appropriate buffer material in the Korean SNF (Spent nuclear fuel) repository site. In this study, hydraulic properties of the WRK compacted bentonite core (4.5 cm in diameter and 1.0 cm in length) as the buffer material were investigated in laboratory experiments. The porosity and the entry pressure of the water saturated core at different confining pressure conditions were measured. The average velocity of water flow in the WRK compacted bentonite core was calculated from results of the breakthrough curves of the CsI aqueous solution and the hydraulic conductivity of the core was also calculated from the continuous flow core experiments. Because various gases could be generated by continuous SNF fission, container corrosion and biochemical reactions in the repository site, the gas migration property in the WRK compacted bentonite core was also investigated in experiments. The gas permeability and the average of gas (H_2) in the core at different water saturation conditions were measured. Laboratory experiments with the WRK Compacted bentonite core were performed under conditions simulating the DGR environment (confining pressure: 1.5-20.0 MPa, injection pressure: 1.0-5.0 MPa, water saturation: 0-100%).

The WRK Compacted bentonite core was saturated at various confining pressure conditions and the porosity ranged from 27.5% to 43.75% (average: 36.75%). The calculated hydraulic conductivity (K) of the core using experimental results was 8.69×10^{-11} cm/s. The gas permeability of the core when the water saturation 0~58 % was ranged of $19.81 \sim 3.43 \times 10^{-16}$ m², representing that the gas migration in the buffer depends directly on the water saturation degree of the buffer medium. The average gas velocity in the core at 58% of water saturation was 9.8×10^{-6} m/s, suggesting that the gas could migrate fast through the buffer medium in the SNF repository site. Identification of the hydraulic property for the buffer medium, acquired through these experimental measurements is very rare and is considered to have high academic values. Experimental results from this study were used as input parameter values for the numerical modeling to simulate the long-term gas migration in the buffer zone and to evaluate the feasibility of the buffer material, controlling the radionuclide-gas migration in the SNF repository site.

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Keywords: Buffer, Compacted bentonite, Engineering barrier, Radionuclide, SNF repository, WRK bentonite

Selection of Design Basis Reference Spent Nuclear Fuels and Safety-Relevant Radionuclides for Geological Repository in South Korea

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Spent nuclear fuel management is a high-priority issue in South Korea, and addressing it is crucial for the country's long-term energy sustainability. The KORAD (Korea Radioactive Waste Agency) is leading a comprehensive, long-term project to develop a safe and effective deep geological repository for spent nuclear fuel disposal. Within this framework, we have three primary objectives in this work.

First, we conducted statistical analysis to assess the inventory of spent nuclear fuel in South Korea as of 2021. We also projected future generation rates of spent nuclear fuels to identify what we refer to as reference spent nuclear fuels. These reference spent nuclear fuels will be used as the design basis spent fuels for evaluating the safety of the repository. Specifically, we identified four types of design basis reference spent nuclear fuels: high and low burnup from PLUS7 (with a 16×16 array) and ACE7 (with a 17×17 array) assemblies.

Second, we analyzed radioactive nuclides' inventory, activities, and decay heats, extending up to a million years after reactor discharge for these reference spent nuclear fuels. This analysis was performed using SCALE/TRITON to generate the burnup libraries and SCALE/ORIGEN for source term evaluation.

Third, to assess the safety resulted from potential radioactive nuclides' release from the disposal canister in future work, we selected safety-related radionuclides based on the ALI (Annual Limit of Intake) specified in Annex 3 of the 2019-10 notification by the NSSC (Nuclear Safety and Security Commission). Conservative assumptions were made regarding annual water intake by humans, canister design lifetime, and aquifer flow rates. A safety margin of 10^{-3} of the ALI was applied. We selected 56 radionuclides that exceed the intake limits and have half-lives longer than one year as the safety-related radionuclides.

However, it is crucial to note that our selection criteria focused on ALI and half-lives. It did not include other essential factors such as solubility limits, distribution coefficients, and leakage processes. So, some of these nuclides can be removed in a specific analysis area depending on their properties.

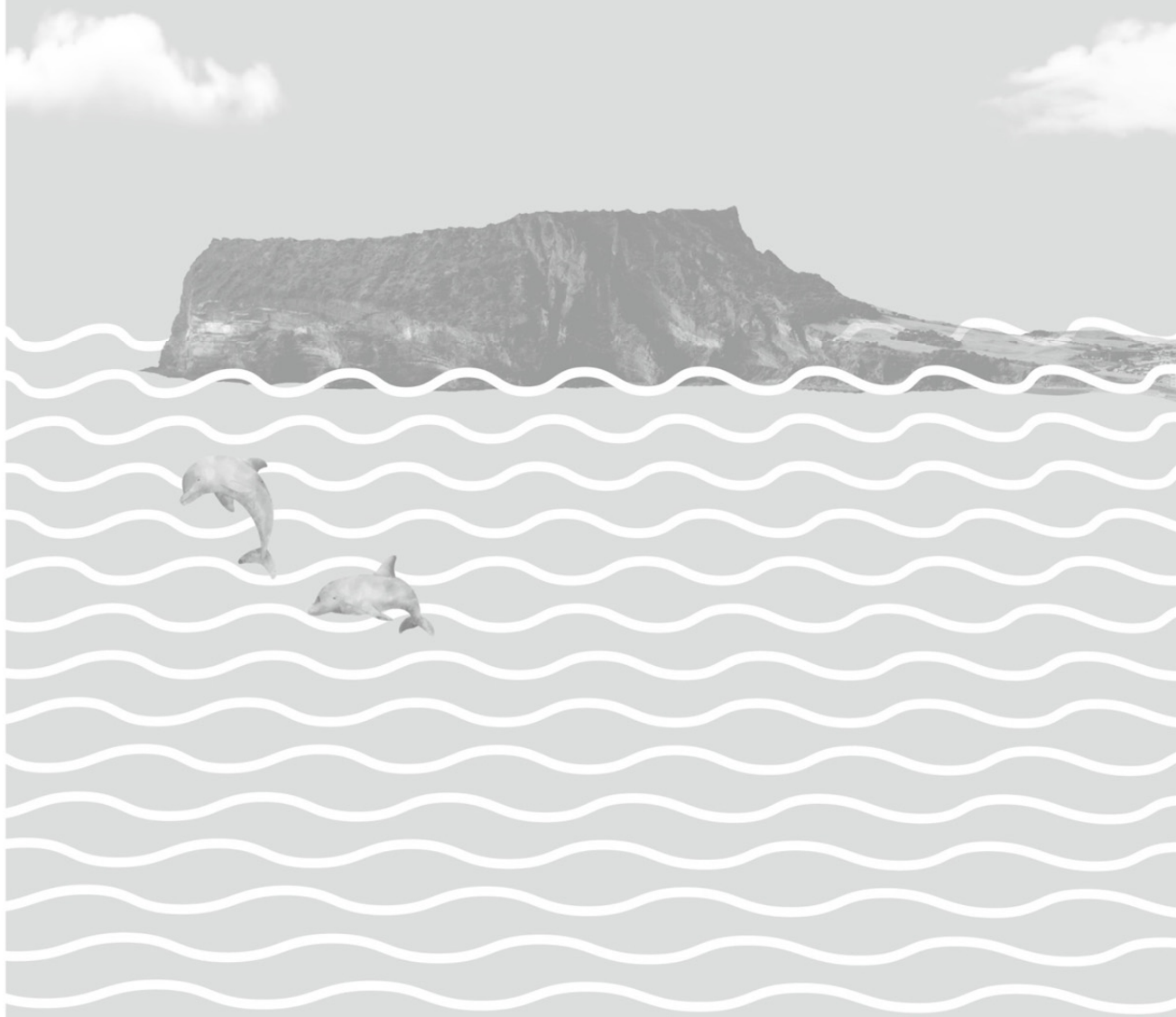
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Keywords: Geological disposal, Safety-relevant radionuclide, Design basis reference spent nuclear fuel, Source terms

3분과

고준위폐기물처분 (POSTER)



In Situ Analysis of Bentonite Buffer Pore Morphology in Groundwater Environments Using Synchrotron Computed Tomography

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This study investigates the behavior of bentonite, used as a buffer material in deep geological disposal systems, in the context of pore morphology under the influence of field-collected groundwater conditions. The bentonite was processed into block form using cold isostatic press (CIP) and subsequently analyzed for its pore morphology in situ using synchrotron X-ray computed tomography (CT) within the field-collected groundwater environment. Bentonite buffers play a critical role in deep geological disposal systems by preventing contact between disposal containers and groundwater. Bentonite typically exhibits swelling upon contact with water, forming few layers of water molecules between its structural layers. However, the presence of ions such as K^+ and Cl^- can lead to a sharp reduction in swelling pressure. Loss of swelling pressure could negatively impact the integrity of future deep geological disposal systems, making its assessment crucial. This study involves processing various types of bentonite, including natural Na-type bentonite, into block forms and subjecting them to exposure in both deionized water and field-collected groundwater conditions. Internal pore morphology changes were measured using X-ray CT technology.

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Keywords: Deep geological disposal, Bentonite, Buffer, Synchrotron X-ray CT

Analysis of Overseas Distribution Coefficients for Radionuclides Sorption on Crystalline Rock Under Aerobic Condition

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A disposal system for spent nuclear fuel mainly divides into two parts; Engineered barriers include spent nuclear fuel, canister, buffer and backfill and natural barriers mean a host rock surrounding engineered barriers. If radionuclides released from a repository, they can migrate to the ecosystem. Sorption plays an important role in retarding the migration of released radionuclides. Hence, the safety assessment for the disposal of a spent nuclear fuel should consider the migration and retardation of radionuclides in geosphere. Distribution coefficient is one of input parameters for the safety assessment. In this work, distribution coefficients for crystalline rock as a natural barrier were collected and evaluated for the purpose of safety assessment for the deep geological disposal of a spent nuclear fuel. The radionuclides considered in this work are 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). The sorption of radionuclides is influenced by various geochemical conditions such as pH/carbonates, redox potential, ionic strength, radionuclide concentration, kinds and amounts of minerals, and microbes. For the evaluation of distribution coefficients, the data from Sweden (SKB), Finland (Posiva), Switzerland (Nagra), and Japan (JAEA) were collected, analyzed, and the recommended distribution coefficients have been suggested.

Keywords: Crystalline rock, Distribution coefficient, Deep-geological disposal, Radionuclide, Safety assessment, Sorption

Thermal Evaluations for Conceptual Design of Deep Geological Repository Considering Improved Modeling of Spent Fuel Decay Heat

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The thermal evaluations for the conceptual design of the deep geological repository considering the improved modeling of the spent fuel decay heat were conducted using COMSOL Multiphysics computational program. The maximum temperature at the surface of a disposal canister for the technical design requirement should not exceed 100°C. However, the peak temperature at the canister surface should not exceed 95°C considering the safety margin of 5°C due to several uncertainties. All thermal evaluations were based on the time-dependent simulation from the emplacement time of the canister to 100,000 years later. In particular, the heat source condition was set to the decay heat rate and axial decay heat profile of the PLUS7 fuel with 4.0wt% U-235 and 45 GWD/MTU. The thermal properties of the granitic rock in South Korea were applied to the host rock region. For the reference design case, the cooling time of the SNF was set to 40 years, the distance between the deposition holes 8 meters and that between the deposition tunnels 30 meters. However, the peak temperature at the canister surface at 10 years was 95.979°C greater than 95°C. This design did not meet the thermal safety requirement and needed to be modified. For the first modified case, when the distance between the deposition tunnels was set to 30 meters, three cooling time cases of 40, 50 and 60 years and five distances of 6, 7, 8, 9 and 10 meters between the deposition holes were considered. The design with the distances of 9 and 10 meters between the deposition holes for the cooling time of 40 years and all five distances for 50 and 60 years were less than 95°C. For the second modified case, when the distance between the deposition holes was set to 8 meters, three cooling time cases of 40, 50 and 60 years and five distances of 20, 25, 30, 35 and 40 meters between the deposition tunnels were considered. The design with the distances of 35 and 40 meters between the deposition tunnels for the cooling time of 40 years, the distances of 25, 30, 35 and 40 meters for 50 years and all five distances for 60 years were less than 95°C. As a result, the peak temperature at the canister surface decreased as the cooling time and the distance between the deposition holes and the tunnels increased.

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Keywords: Thermal evaluation, Disposal canister, Deposition hole, Deposition tunnel, Spent nuclear fuel, COMSOL multiphysics program

Morphological Observation of the Corrosion of Double Corrosion Resistant Barrier

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In this study, a third metal layer with a higher corrosion potential than copper was introduced between the copper and cast iron layer to strengthen the corrosion resistance of the copper layer which is considered as a corrosion resistant barrier in the disposal container for spent nuclear fuel.

Three types of corrosion-resistant metals, silver, nickel, and titanium, were selected as the intermediate insertion layer, and the galvanic specimens of two bonded metals were exposed to KURT (KAERI Underground Research Tunnel) groundwater and a high voltage of 1.0 V was applied to corrode the specimens at electrochemically accelerated condition. Corrosion of copper part was confirmed in Cu-Ti, Cu-Ni, and Cu-Ag galvanic specimens, but copper part was not corroded in Cu-Fe galvanic specimen.

If the corrosion-resistant intermediate layer proposed in this study works properly, the local corrosion problem of copper disposal canister is expected to be some degree solved, which can apply to a welding part or a stress concentrated part.

Keywords: Copper, Galvanic corrosion, Disposal canister, Cold spray coating, Double layer coating

A GoldSim Implementation of Safety Assessment Model for a Double-Layered SF Repository

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Since the first operation of the Gori No. 1 nuclear power plant in Korea was started to operate in 1978, currently 24 nuclear power plants have been being operated, out of which 21 plants are PWR types and the rest are CANDU types. About 30% of total electricity consumed in Korea is from all these nuclear power plants. The accumulated spent nuclear fuels (SFs) generated from each site are temporarily being stored as wet or dry storage type at each plant site. These SFs with their high radiotoxicity, heat generating, and long-lived radioactivity are actually the only type of high-level radioactive waste (HLW) in Korea, which urgently requires to be disposed of in deep geological repository. Studies on disposal of HLW in various kind of geological repositories have been carried out in such countries as Sweden, Finland, United States, and etc. with their own methodologies and management policies in consideration of their situations. In Korea long-term R&D research program for safe management of SF has also been conducted during last couple of decades since around 1997, during which several various alternative type of disposal concepts for disposal of SNFs in deep geological formations have been investigated and developed. The first concept developed was KAERI Reference Disposal System (KRS) which is actually very much similar to Swedish KBS-3, a famous concept of direct disposal of SF in stable crystalline rock at a depth of around 500 m which has been regarded as one of the most plausible method worldwide. The world first Finnish repository which is expected to begin to operate sooner or later will be also this type. Since the characteristics of SF discharged from domestic nuclear reactors have been changed and improved, and burnup has sometimes increased, a more advanced deep geological repository system has been needed, KRS-HB (KRS with High Burnup SF) has been developed and in consideration of the dimensions of SNFs and the cooling period at the time point of the disposal time, KRS+, a rather improved disposal concept has also been subsequently developed which is especially focused on the efficient disposal area. Recently research has concentrated on rather advanced disposal technology focused on a safer and more economical repository system in recent view of the rapidly growing amount of accumulated SF. Especially in Korea the rock mass and the footprint area for the repository extremely limited for disposal site. Some preliminary studies to achieve rather higher efficiency repository concept for disposal of SF recently have already been emphasized. Among many possible ones for consideration of design for high-efficiency repository system, a double-layered system has been focused which is expected to maximize disposal capacity within the minimum footprint disposal area. Based on such disposal strategy a rather newly designed performance assessment methodology might be required to show long-term safety of the repository. Through the study some prerequisites for such methodological development has been being roughly checked and investigated, which covers FEP identification and pathway and scenario analyses as well as preliminary conceptual modeling for the nuclide release and transport in near-field, far-field, and even biosphere in and around the conceptual repository system. Through the study such scenarios and models has been implemented to development of a safety assessment by utilizing GoldSim development tool for a rough quantitative comparison with existing disposal options and simple illustration purpose as well as for showing how to develop and implementation of the model to GoldSim templet.

Keywords: Double-layered repository, Spent fuel disposal, Safety assessment, GoldSim modeling

Effective Retention of Anionic Iodine Within a Compacted Bentonite

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The spent fuels derived from the nuclear reactor facilities may be finally disposed in a deep underground below 500 m. It majorly has uranium with minor iodine, which is a typical anionic radionuclide. In particular, radioiodine has higher mobility from its spent fuel source. It has been well known that it could freely pass through a compacted bentonite that is one of underground engineering barriers that are used to retard some nuclide's migration from the spent fuel. We installed a small laboratory apparatus in an anaerobic glove box imitating such an underground repository and evaluated the iodine mobility in compacted bentonites with or without copper. Some copper-bearing bentonites were prepared in two types, a copper ion-exchanged form and a copper nanoparticle-mixed one. In our study, we tried to find an effect of sulfate that has an ability to retard mobile iodine from the compacted bentonite for a long-term period.

Conclusively, we found an effective way to limit the iodine release from the compacted bentonite. This condition can be achievable by exchanging the bentonite interlayer cations with copper ions or by simply mixing copper nanoparticles with bentonite powder. In those cases, soluble iodine can be easily immobilized as a solid phase (i.e., marshite (CuI)) by combining with copper via the geochemical role of sulfate and indigenous SRB (sulfate reducing bacteria) of bentonite.

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Keywords: Iodine, Compacted bentonite, Copper, Sulfate

Alternative Conceptual Design of the Disposal System for CANDU Spent Nuclear Fuels in Korea

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Even though a huge amount of spent nuclear fuels are accumulated at each nuclear power plant site in Korea, our government has not yet started to select a final disposal site, which might require more than several km² surface area. According to the second national plan for the management of high-level radioactive waste, the reference geological disposal concept followed the Finnish concept based on KBS-3 type. However, the second national plan also mentioned that it was necessary to develop the technical alternatives. Considering the limited area of the Korean peninsula, the authors had developed an alternative disposal concepts for spent nuclear fuels in order to enhance the disposal density since 2021. Among ten disposal concepts shown in the literature published in 2000's, we narrowed them to four concepts by international experiences and expert judgements.

Assuming 10,000 t of CANDU spent nuclear fuels (SNF), we designed the engineered barriers for each alternative disposal concept. That is, using a KURT geological conditions, the engineered barrier systems (EBS) for the following four alternative concepts were proposed: ① mined deep borehole matrix, ② sub-seabed disposal, ③ deep borehole disposal, and ④ multi-level disposal. The quantitative data of each design such as foot prints, safety factors, economical factors are produced from the conceptual designs of the engineered barriers. Five evaluation criteria (public acceptance, safety, cost, technology readiness level, environmental friendliness) were chosen for the comparison of alternatives, and supporting indicators that can be evaluated quantitatively were derived. The AHP with domestic experts was applied to the comparison of alternatives. The two-level disposal was proposed as the most appropriate alternative for the enhancement of disposal efficiency by the experts. If perspectives changes, the other alternatives would be preferred.

Three kinds of the two-level disposal of CANDU SNF were compared. It was decided to dispose of all the CANDU spent nuclear fuels into the disposal holes in the lower-level disposal tunnels because total footprint of the disposal system for CANDU SNF was much smaller than that for PWR SNF. Currently, we reviewed the performance criteria related to the disposal canister and the buffer and designed the EBS for CANDU SNF. With the design, safety assessment and cost estimates for the alternative disposal system will be carried out next year.

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Keywords: CANDU spent nuclear fuel, Alternative conceptual design, Engineered barrier, Disposal canister, Multi-level disposal

Piping Erosion and Cracking Behavior of Compacted Bentonite Upon Technological Voids Filling

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The presence of technological voids in deep geological repositories for high-level radioactive nuclear waste can have negative effects on the hydro-mechanical properties of the engineered barrier system when groundwater infiltrates from the surrounding rock. This study conducted hydration tests along with image acquisition and X-ray CT analysis on compacted Korean bentonite samples, which simulated technological voids filling to investigate the behavior of fracturing (piping erosion) and cracking deterioration. We utilized a dual syringe pump to inject water into a cell consisting of a bentonite block and technological voids at a consistent flow rate. The results showed that water inflow to fill technological voids led to partial hydration and self-sealing, followed by the formation of an erosional piping channel along the wetting front. After the piping channel generated, the cyclic filling-piping stage is characterized by the repetitive accumulation and drop of water pressure, accompanied by the opening and closing of piping channels. The stoppage of water inflow leads to the formation of macro- and micro cracks in bentonite due to moisture migration caused by high suction pressure. These cracks create preferential flow paths that promote long-term groundwater infiltration. The experimental test and analysis are currently ongoing. Further experiments will be conducted to investigate the effects of different dry density in bentonite, flow rate, and chemical composition of injected water.

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Keywords: Technical voids, Piping erosion, Self-sealing, Bentonite buffer, X-ray CT

Comparison of Bentonite Candidates for Korea Disposal System

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Buffer materials are one of the engineering barrier components in high-level radioactive waste disposal facilities. Compacted bentonite has been known as the most suitable buffer material so far, and research is being conducted worldwide to determine the characteristics of suitable bentonite blocks in each country. Therefore, this study aims to compare and analyze various properties of different buffer material candidates, including thermal, hydraulic, and mechanical properties. Buffer material candidates for Korea disposal system, Kyungju Bentonite (KJ-I, KJ-II), and Bentonil-WRK were compared. The properties were compared and analyzed based on experimental and literature data. The data obtained from this report can be used to understand the behavior of buffer materials and assess whether they meet various criteria, such as temperature and saturation, and ultimately serve as an essential input variable database for safety evaluations of disposal systems.

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Keywords: Bentonite, High-level nuclear waste disposal, Properties, Candidates

Experimental Investigation of Radionuclide Transport by Gas Bubbles in a Deep Disposal Repository

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The final disposal of Spent Nuclear Fuel (SNF) will take place in a deep geological repository. The metal canister surrounding the SNF is made of cast iron and copper, designed to provide long-term containment of radionuclides. Canister is intended to be safeguarded by a multiple-barrier disposal system comprising engineered and natural barriers. Colloids and gases are mediators that can accelerate radionuclide migration and influence radionuclide behavior when radionuclides leak from the canister at the end of its service life. It is very important to consider these factors in the assessment of the long-term stability of deep disposal repository. An experimental setup was designed to observe the acceleration of nuclide behavior due to gas-mediated transport in a simulated environment with specific temperature and pressure conditions, similar to those of a deep disposal repository.

In this study, experiments were conducted to simulate gas flow within an engineered barrier under conditions reflecting 1000 years post repository closure. The experiment utilized bentonite WRK with a dry density of 1.61 g/cm³ after compaction. The compacted bentonite was subsequently saturated under a water pressure of 5 MPa, equivalent to the hydrostatic pressure found 500 meters underground. Gas was introduced into the saturated bentonite at different pressures to assess the permeation behavior of the bentonite relative to gas pressure variations. Consequently, it was observed that under specific pressures, gas permeated the saturated bentonite, ascending in the form of bubbles. Furthermore, it was noted that when a continuous flow was initiated within the bentonite, erosion took place, leading to the buoyant transportation of eroded particles upward by the bubbles. The particles transported by the bubbles had a relatively small particle size distribution, and cesium also tended to be transported by the bubbles and moved upward. When high-pressure gas is generated at the interface of the canister and the buffer, flow through the buffer can occur, and cationic nuclides such as cesium and strontium can be attached to the gas bubble and migrate. However, the pressure of the gas to break through the saturated buffer is very high, and the amount of cesium transported by the gas bubbles is very limited.

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Keywords: Nuclide behavior, Gas breakthrough, Accelerated nuclide transport, Bentonite, Deep geological repository

Mineralogical Characteristics of Na- and Ca-bentonite

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Buffer materials play an important role in preventing the leakage of radionuclides from the residue. The mineralogical properties of these buffer materials are critical in repository design. This study presents the fundamental properties of Na-type MX80 and a novel Ca-type Bentonil-WRK. The CaO to MgO ratio in Bentonil-WRK was approximately 1:1, and the CaO to Na₂O ratio was approximately 2.8:1. These results suggest that Bentonil-WRK demonstrates a lower swelling index compared to Gyeongju bentonite due to its CaO-to-MgO ratio's proximity to 1:1, despite having a higher montmorillonite content than Gyeongju bentonite. The results of this research can provide useful foundational data for the evaluation of the thermal-hydraulic-mechanical-chemical behavior of buffer materials.

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Keywords: Buffer material, Ca-type bentonite, Na-type bentonite, Mineralogical analysis, Geological deep disposal

Intrusion and Erosion of Bentonite Buffer Materials Considering Hydro-Mechanical-Chemical Effects

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The compacted bentonite buffer is a key component of the engineered barrier system in deep geological repositories for high-level radioactive waste disposal. Groundwater infiltration into the deep geological repository leads to the saturation of the bentonite buffer. Bentonite saturation results in bentonite swelling, gelation and intrusion into the nearby rock discontinuities within the excavation damaged zone of the adjacent rock mass. Groundwater flow can result in the erosion and transport of bentonite colloids, resulting in bentonite mass loss which can negatively impact the long-term integrity and safety of the overall engineered barrier system. The hydro-mechanical-chemical interactions between the buffer, surrounding host rock and groundwater influence the erosion characteristics of the bentonite buffer. Hence, assessing the critical hydro-mechanical-chemical factors that negatively affect bentonite erosion is crucial for the safety design of the deep geological repository.

In this study, the effects of initial bentonite density, aperture, discontinuity angle and groundwater chemistry on the erosion characteristics of Bentonite WRK are investigated via bentonite extrusion and artificial fracture experiments. Both experiments examine bentonite swelling and intrusion into simulated rock discontinuities; cylindrical holes for bentonite extrusion experiments and plane surfaces for artificial fracture experiments. Compacted bentonite blocks and bentonite pellets are manufactured using a compaction press and granulation compactor respectively and installed in the transparent extrusion cells and artificial fracture cells. The reference test condition is set to be 1.6 g/cm³ dry density and saturation using distilled water. After distilled water or solution injection, the axial and radial expansion of the bentonite specimens into the simulated rock discontinuities are monitored for one month under free swelling conditions with no groundwater flow. Subsequent flow tests are conducted using the artificial fracture cell to determine the critical flow rate for bentonite erosion. The intrusion and erosion characteristics are modelled using a modified hydro-mechanical-chemical coupled dynamic bentonite diffusion model and a fluid-based hydro-mechanical penetration model.

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Keywords: Engineered barrier system, Bentonite buffer, Bentonite erosion, Artificial fracture test, Numerical modelling

Formation, Distribution, and Characterization of Zeolite in Pohang Basin: A Natural Analogue Study

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Natural analogue studies aim to understand specific processes or to make predictions regarding formation processes or environmental conditions based on information from natural phenomena on Earth. Studying geological environments similar to the disposal system can help evaluate the long-term stability of the disposal system. These studies play a crucial role in evaluating the long-term stability of deep geological repositories (DGRs) for high-level radioactive wastes, which must remain stable for extended periods. In particular, bentonite, as a vital buffer material in DGRs, is mainly composed of montmorillonite and undergoes a transformation into zeolite under specific environmental conditions, resulting in the loss of its role as a buffer material in terms of swelling property and hydraulic conductivity. In this study, bentonite samples from Pohang Basin in Korea were investigated, revealing the presence of montmorillonite and zeolite in both outcrop and drilling core samples. While it has been known that montmorillonite and zeolites can form from volcanic glass through diagenesis or hydrothermal alteration, this study revealed that these minerals are formed under distinct environmental conditions. The outcrop samples primarily consisted of montmorillonite with minor quantities of cristobalite and amorphous silicate minerals. In contrast, the drilling core samples were composed of montmorillonite, zeolites, quartz and feldspar, indicating different mineral assemblages and characteristics between the outcrop and drilling samples. This observation suggests different environmental conditions during the mineral formation process. Therefore, various experiments and analytical methods were employed to gain a deeper understanding of the phase transformation from montmorillonite to zeolites under diverse environmental conditions.

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Keywords: Natural analogue, Montmorillonite, Zeolite, High-level radioactive waste disposal

Thermodynamic Modeling of Iodide and Benzoate Sorption Onto Ca-type Bentonil-WRK Montmorillonite

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The permanent disposal of discharged spent nuclear fuel (SNF) and contaminated radioactive waste generated from the subsequent chemical treatments of SNF has become a serious pending issue in many countries that operate the nuclear power plants. Among the diverse engineering solutions proposed for the disposal of high-level radioactive waste (HLW), deep geological disposal (DGD) has been considered as the most proven and safe option to prevent any significant release of radionuclides into the biosphere and to predictably ensure the long-term performance of disposal system. The DGD system consists of multiple structural components; the bentonite clay-based buffer and tunnel backfills are designed to perform the primary hydrogeochemical functions of 1) inhibiting the ingress of groundwater and reactive substances that could compromise the integrity of canister and 2) retarding the migration of released radionuclides into biosphere by providing the sufficient chemisorption sites. Montmorillonite, which is a 2:1 phyllosilicate mineral belonging to smectite group, constitutes the majority of bentonite, and it mainly predominate the swelling and chemisorption capacities of the clay material. Thus, it is essentially required to thoroughly understand the chemical interactions of major radionuclides and other important substances with montmorillonite in advance to accurately evaluate the long-term retention performance of engineered barriers and to reduce the uncertainties in the safety assessment of a deep geological repository (DGR) ultimately. Thus far, sorption of dissolved species onto mineral adsorbents has been generally described and quantified using the simple sorption-desorption distribution coefficient (K_d) concept; since any specific reaction mechanisms are not considered and reflected in the K_d concept, an empirical K_d value is intrinsically dependent on the aqueous conditions under which it was measured. In this framework, substantial scientific efforts have been made to develop a robust basis for geochemically parametrizing the sorption phenomena more reliably, and the application of thermodynamic sorption modeling (TSM), which is based on the chemical principle of mass action laws, has been studied with the aim of improving overall confidence in the description of radionuclide migration under a wide range of aquatic conditions. The disposal performance demonstration R&D division of KAERI introduced a new reference Ca-bentonite clay called Bentonil-WRK (Clariant Korea) for HLW disposal research in 2021 as the domestic Ca-bentonite sources have been depleted. We successfully separated and purified Ca-montmorillonite from the Bentonil-WRK clay, and its geochemical characteristics were meticulously studied by means of XRD, BET, CEC, FT-IR analyses and controlled acid-base titration. In this work, chemical sorption behaviors of aqueous iodide and benzoate, which are a major fission product in HLW and a model ligand of complex natural organic matters present in the deep geological environment, onto the purified Ca-montmorillonite were assessed under ambient conditions of $S/L = 5$ g/L, $I = 0.01$ M CaCl_2 , $\text{pH} = 4\text{--}9$, $\text{pCO}_2 = 10^{-3.4}$ atm, and $T = 25^\circ\text{C}$. Further, their unique adsorption envelopes and corresponding thermodynamic reaction constants refined from the diffuse double layer model (DDLDM)-based inverse modeling of experimental sorption data were discussed.

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Keywords: Radionuclide migration, Engineered barrier, Montmorillonite, Iodide, Benzoic acid, Sorption, Thermodynamic modeling

Characterization of Indigenous Bacterial Communities in Bedrock Aquifer: Toward a Safe Deep Geological Repository System of High-Level Radioactive Wastes

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This study aimed to provide better understanding of the bedrock aquifer bacterial communities and their functions in deep geological repository (DGR) environment. Two study sites of uranium deposits in the Ogcheon Metamorphic Belt were selected: Boeun and Guemsan. From two study sites, six groundwater samples were obtained with different boreholes and depths: OB1 (Boeun, 25 m), OB3 (Boeun, 80 m), GS1 (Guemsan, 25 m), GS2 (Guemsan, 85-90 m), GS3-I (Guemsan, 32-38 m), GS3-II (Guemsan, 70-74 m). The physicochemical properties of groundwater were analyzed by multi-parameter sensors, ion chromatography (IC), and inductively coupled plasma optical emission spectroscopy (ICP-OES). Illumina Miseq sequencing was performed to investigate bacterial community in six groundwater samples. In addition, the number of sulfate-reducing bacteria (SRB) was quantified by a quantitative PCR (qPCR). Bacterial community composition varied in response to boreholes and depths. A total of 14 different phyla and 36 classes were detected from six groundwater samples. Overall, *Proteobacteria*, *Actinomycetota*, and *Bacteroidota* were dominant in the phylum level. SRB and iron-reducing bacteria (IRB) were detected in all groundwater samples even though organic carbon sources were not abundant (0.7-3.3 mg-total organic carbon/L). This result shows a potential to immobilize uranium in DGR environment. In particular, SRB, *Desulfosporosinus fructosivorans* and *Humidesulfovibrio mexicanus* were mainly detected in GS1 and GS2 groundwater samples, which attributed to higher dissimilatory sulfite reductase functional gene copy number in GS1 and GS2 groundwater samples. Statistical analysis was performed to understand the correlation between environmental factors and core bacterial species. Dissolved oxygen (DO), Fe, and Mn concentrations were positively correlated with *Curvibacter fontanus* while *Undibacterium rivi* had a negative correlation with pH. These results indicate that bacterial community could be changed in response to environmental variation. Further study with a greater number of samples is necessary to obtain statistically reliable and meaningful results for a safe DGR system.

Keywords: Illumina miseq, Bacterial community, Sulfate-reducing bacteria (SRB), Iron-reducing bacteria (IRB), qPCR

Review on the Geochemical Effects on the Coupled Processes of Discontinuous Natural Barrier System

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The natural barrier system surrounding the geological repository for high-level radioactive waste plays a crucial role in preventing or delaying the leakage of radionuclides. Therefore, the natural barrier should ensure low permeability to prevent groundwater flow into the engineered barrier system throughout the repository's lifetime. Crystalline rock, considered as the host rock for the geological repository in Korea, exhibits low intact rock permeability, but the crystalline rock often contains the multiple discontinuities due to its high brittleness that can allow the unexpected fluid flow. Therefore, the long-term hydraulic behavior of the discontinuity should be characterized while considering additional thermal, mechanical, and chemical effects. In comparison to thermal, hydraulic, and mechanical processes, the chemical processes on the discontinuities progress relatively slowly, resulting in limited researches to include these chemical processes.

This research introduces mechanisms the involving coupled thermal-hydraulic-mechanical-chemical processes focusing on the rough fracture surfaces and asperities. The chemically-induced changes in mechanical and hydraulic properties are described based on pressure solution and precipitation concepts. A comprehensive review of laboratory tests, field tests, and numerical simulations is conducted related to the chemically-induced coupled processes in fractured rock. Laboratory tests, in particular, concentrate on microscopic changes in fracture asperities induced by pressure solution to analyze chemically-induced aperture changes. The TOUGHREACT, an integral finite difference method program for thermal-hydraulic-chemical simulations, is generally employed to model the chemical response of pressure solution and precipitation on fracture surfaces. The TOUGHREACT includes a module to describe effective porosity and permeability changes based on the modified cubic law, so the real-time change of the fracture permeability can be reflected during the flow simulation. Considering the coupled thermal-hydraulic-mechanical-chemical processes of discontinuity, it becomes evident that the chemical processes under repository conditions (long-term, high temperature, and high pressure) can disturb the hydraulic performance of the natural barrier, so further research is required to characterize the chemically-induced coupled processes for assessing the long-term performance of the natural barrier system.

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Keywords: Coupled processes, Natural barrier system, Fractured rock, Dissolution

Numerical Analysis of the Barcelona Basic Model by the Discontinuum-Based Method

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The Barcelona Basic Model (BBM) is an elasto-plastic model used to describe the coupled thermo-hydro-mechanical behaviors of unsaturated soil. BBM is frequently adopted to model the unique swelling behavior of bentonite, which is generally considered as the buffer material between the host rock and the canister containing high-level radioactive waste in deep geological repositories, under the changing thermal, hydraulic, mechanical and chemical conditions during the lifetime of repository. Therefore, a variety of the continuum-based numerical methods tried to add the BBM for modelling the multi barrier systems of geological repository and succeeded to describe the elasto-plastic deformation of bentonite. However, to demonstrate the entire barrier systems the host rock should be modelled simultaneously with the buffer materials, and the continuum-based methods may be limited in their ability to reflect the fracture networks in the host rock which could be the major flow channels of groundwater. This research applies BBM in 3DEC, a three-dimensional block-based discrete element method, and validates the model by comparing the change of specific volume and mean effective stress during three numerical test cases. Discontinuum-based numerical methods with BBM can be extended to describe the coupled thermo-hydro-mechanical processes of multi-barrier systems in geological repositories, with a focus on the interaction between the host rock and bentonite.

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Keywords: Numerical simulation, Barcelona basic model, Bentonite, Discontinuum, Geological repository

Geochemical Behaviors of Uranium in Groundwater Samples From Natural Analogue Study Sites in Ogcheon Metamorphic Belt

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The mobility of uranium (U) in various disposal environments of a deep geological repository is controlled by various geochemical conditions and parameters. In particular, oxidation state of uranium is considered as a major factor to control the mobility of uranium in most of geological environments. In this study, therefore, we investigated the geochemical behaviors of uranium in groundwater samples from natural analogue study sites located in the Ogcheon Metamorphic Belt (OMB). Groundwater samples were taken using a packer system from Boeun Hoenam-myun site and Geumsan Suyoung-ri site where several boreholes were drilled with various depths. The geochemical properties and parameters such as temperature, pH, Eh, EC, and DO were directly measured in the site using an in-line measurement method. The concentrations of major cations and anions in the groundwater samples were measured by using ICP-OES (Inductively Coupled Plasma-Optical Emission Spectrometry) and IC (Ion Chromatography), respectively. The concentrations of trace elements including U and Th were measured by using ICP-MS (Inductively Coupled Plasma-Mass Spectrometry). The concentrations of U in the groundwater samples are very low for the Hoenam-myun site (0.03~0.69 ppb) and Suyoung-ro site (0.39~1.74 ppb) even though the two sites are uranium deposits and redox conditions are weakly oxidizing. The speciation, saturation index (SI), pH-Eh (Pourbaix) diagram were calculated using the Geochemist's Workbench (GWB 9.0) program and the recent OECD/NEA thermochemical database for U. Calculation results for U speciation in the groundwater samples show that major dissolved uranium species in the groundwater samples are mainly as calcium uranyl carbonate complexes such as $\text{Ca}_2\text{UO}_2(\text{CO}_3)_3(\text{aq})$ and $\text{CaUO}_2(\text{CO}_3)_3^{2-}$ for almost all groundwater samples. The calculated results for SI and Pourbaix diagram also show that the dominant uranium solid phase is a uranyl silicate mineral, uranophane ($\text{Ca}(\text{H}_2\text{O})(\text{U}^{\text{VI}}\text{O}_2)_2(\text{SiO}_2)_2(\text{OH})_6$), not uraninite ($\text{U}^{\text{IV}}\text{O}_2$). Since the determination of Eh values for natural groundwater samples is very difficult and uncertain work, we analyzed and discussed the effect of Eh on the geochemical behaviors of U in the groundwater. However, these calculation results are not consistent with the observation for U minerals in rock samples using electron microscopic techniques. Thus, we need further studies to explain the discrepancy between calculation and observation results.

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Keywords: Natural analogue study, Uranium deposit, Groundwater, Uranium speciation, Uranium mineral

Safety Function, Performance Targets and Technical Requirements for the EBS System of an Alternative Repository Concept for SNFs in Korea

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According to the second high-level radioactive waste management national basic plan announced in December 2021, the reference geological disposal concept for spent nuclear fuels (SNF) in Korea followed the Finnish concept based on KBS-3 type. Also, the basic plan required consideration of the development of the technical alternatives. Accordingly, Korea Atomic Energy Research Institute is conducting analyses of various alternative disposal concepts for spent nuclear fuels and is in the final selection stage of an alternative disposal concept. 10 disposal concepts including reference concept were considered for analysis in terms of disposal efficiency and safety. They were reference concept, mined deep borehole matrix, sub-seabed disposal, deep borehole disposal, multi-level disposal, space disposal, sub-sea bed disposal, long-term storage, deep horizontal borehole disposal, and ice-sheet disposal. Among them, first 4 concepts, mined deep borehole matrix, sub-seabed disposal, deep borehole disposal, multi-level disposal, were selected as candidate alternative disposal concepts by the evaluation of qualitative items. And then, by the evaluation of quantitative and qualitative items with specialists, multi-level disposal concept was being selected as a final alternative disposal concept.

Design basis and performance requirements for designing alternative disposal systems were laid in the previous stage. Based on this, the design strategy and main design requirements were derived, and the engineered barrier system of a high-efficiency disposal concept was preliminary designed accordingly. In addition, as an alternative disposal concept, performance targets and related requirements were established to ensure that the high-efficiency repository system and its engineered barrier system components, such as disposal containers, buffer bentonites, and backfill perform the safety functions.

Items that qualitatively describe safety functions, performance goals, and related requirements at this stage and items whose quantitative values are changed according to future test results will be determined and updated in the process of finalizing and specifically designing an alternative high-efficiency disposal system.

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Keywords: Safety function, Performance targets, Technical requirements, Spent nuclear fuel, Alternative disposal, Engineered barrier system, Multi-level disposal

Application of Outlier Detection Algorithms to Design a Multi-Packer System for Hydrogeological Analyses

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The high-level radioactive waste repository must ensure its performance for a long period of time enough to sufficiently reduce the potential risk of the waste, and for this purpose, multi-barrier systems consisting of engineered and natural barrier systems are applied. If waste nuclides leak, the dominating mechanisms facilitating their movement toward human habitats include advection, dispersion and diffusion along groundwater flows. Therefore, it is of great importance to accurately assess the hydrogeological and geochemical characteristics of the host rock because it acts as a flow medium. Normally, borehole investigations were used to evaluate the characteristics and the use of multi-packer system is more efficient and economical compared to standpipes, as it divides a single borehole into multiple sections by installing multiple packers. For effective analyses and groundwater sampling, the entire system is designed by preselecting sections where groundwater flow is clearly remarkable.

The selection is based on the analyses of various borehole and rock core logging data. Generally, sections with a high frequency of joints and evident water flow are chosen. Analyzing the logging data, which can be considered continuous, gives several local points where the results exhibit significant local changes. These clear deviations can be considered outliers within the data set, and machine learning algorithms have been frequently applied to classify them. The algorithms applied in this study include DBSCAN (density based spatial clustering of application with noise), OCSVM (one class support vector method), KNN (K nearest neighbor), and isolation forest, of which are widely used in many applications. This paper aims to evaluate the applicability of the aforementioned four algorithms to the design of multi-packer system. The data used for this evaluation were obtained from DB-2 borehole logging data, which is a deep borehole locates near KURT.

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Keywords: Multi-packer system, Outlier detection, Machine learning, Borehole logging, Joint distribution

Post-Processing Approach for Massive Output Data From Large-Scale TSPA in APro

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APro, a process-based total system performance assessment (TSPA) tool for a geological disposal system, has a framework for simulating the radionuclide transport affected by thermal, hydraulic, mechanical or geochemical changes occurred in the disposal system. APro aims to be applied for the TSPA to long-term ($> 100,000$) evolution scenarios in real-world repository having more than 10,000 boreholes. In this large-scale TSPA, it is important not only to develop a high-performance numerical approach, but also to apply an efficient post-processing approach to massive spatio-temporal data.

The post-processing refers to validating numerical analysis results, analyzing and evaluating target systems through data processing or visualization. Since APro uses COMSOL interface, the post-processing function in COMSOL can be used. However, when the data size increases due to large-scale numerical analysis, the time for the COMSOL post-processing increases, resulting in a problem that the analysis and evaluation are not performed effectively. In this case, it is possible to extract necessary data using the COMSOL exporting function and importing it into an external post-processing program for the analysis and evaluation.

In this study, the efficiency of external post-processing with extracted data from COMSOL was reviewed. And, we derived a proper data extraction approach (format and structure) that can increase efficiency of external post-processing.

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Keywords: APro, Total system performance assessment model, Post-processing

Differences in Copper Corrosion Rate Depending on Manufacturing Method in Domestic Groundwater Concentration

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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 Korea, copper and iron are being considered for external and internal canisters, and bentonite is being considered as a buffer material. This is a similar choice to many overseas disposal systems. However, unlike the rolling, extrusion, and forging manufacturing methods being considered overseas for manufacturing external canister, domestic research is currently underway on manufacturing external copper canister using cold spray coating. The canister manufacturing method may vary depending on unit cost and manufacturing convenience. However, the properties of metal vary slightly depending on the manufacturing method of the metal. In this case, the characteristics of the canister may vary slightly depending on the canister manufacturing method, and eventually the corrosion resistance may also vary slightly. In order to understand how the copper canister manufacturing method affects corrosion resistance, corrosion rates were calculated and compared through electrochemical corrosion experiments at domestic groundwater ion concentration.

Keywords: Copper, Canister, Corrosion, Groundwater, Manufacturing method

Investigating Microstructure in Copper Canister Welds for Deep Geological Disposal via Wire Arc Additive Manufacturing

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Due to the necessity of isolating spent nuclear fuel (SNF) from the human life zone for a minimum of 10^6 years, deep geological disposal (DGD) has emerged as a prominent solution for SNF management in numerous countries. Consequently, the resilience of disposal canisters to corrosion over such an extended storage period becomes paramount. While copper exhibits a relatively low corrosion rate, typically measured in millimeters per million years, in geological environment, special attention must be directed towards verifying the corrosion resistance of copper canister welds. This validation becomes inevitable during the sealing of the disposal canister once SNFs are loaded, primarily because the weld zone presents a discontinuous microstructure, which can accelerate both uniform and localized corrosion processes. In this research, we conducted an in-depth analysis of the microstructural characteristics of copper welds manufactured by TIG-based wire arc additive manufacturing, which is ideal for welding relatively large structures such as a disposal canister. To simulate the welds of copper canister, a 12 mm thick oxygen-free plate was prepared and Y and V grooves were applied to perform overlay welding. Both copper welding zones were very uniform, with negligible defects (i.e., void and cracks), and contained relatively large grains with columnar structure regardless of groove types. For improving microstructures at welds with better corrosion resistance, the effect of preheat temperature also investigated up to 600°C.

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Keywords: Disposal canister, Copper welds, Deep geological disposal, Wire arc additive manufacturing

~~Geological Risk Assessment for Deep Underground Tunneling in Korea (KGRAUT)~~

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Withdrawal

Long-Term Experimental Study on Concrete Deterioration as a Natural Analogue Study of Radioactive Waste Disposal

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In Natural Analogue Study, Concrete is one of the important engineering barrier components in the Multi-thin wall concept of radioactive waste disposal and plays the most important role in disposal sites. The concrete barrier at the disposal site loses its role as a barrier due to various deterioration phenomena such as settlement, earthquake, and ground movement, causing the disposed waste to leak into the natural ecosystem.

Some of the key factor is deterioration triggered by sulfate attack. Concrete deterioration induced by sulfate is commonly manifested in an extensive scale when a concrete structure makes contact with soil or water, aggravating its performance.

In this study, an accelerated concrete deterioration evaluation experiment was performed using a total of three experimental methods to evaluate the reaction between concrete and water.

The first experiment was a deterioration evaluation using Demi. Water, the second was a deterioration evaluation using KURT groundwater after extraction, and the last experiment was a concrete deterioration evaluation using KURT groundwater and sodium sulfate.

For all of these experiments, accelerated concrete deterioration experiments were conducted after immersion for a total of 365 days, and specimens were taken out at 30-day intervals and characterization analysis such as SEM and EDS was performed. Experimental analyzes have shown that various chemical species are generated or destroyed over time.

In the future, we plan to continue to conduct a total of three concrete deterioration evaluation experiments above, and additionally evaluate the chemical reaction between bentonite and concrete.

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Keywords: Concrete, Deterioration, Natural analogue study, Sulfate attack

Thermal Analysis of Engineered Barrier System for Long-Term Safety of a Deep Geological Repository

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In order to ensure the long-term safety of a deep geological repository, the performance assessment of the Engineered Barrier System (EBS) considering a thermal process should be performed. The maximum temperature at the side wall of a disposal canister for the technical design requirement should not exceed 100°C. In this study, the thermal modelling was conducted to analyze the effects of the thermal process from a disposal canister to the surrounding near-field host rock using the PFLOTRAN code. The mesh was generated using the LaGriT code and the material properties were assigned by applying the FracMan code. Initial conditions were set as the average geothermal gradient (25.7°C/km) and an average surface temperature (14.7°C) in Korea. The highest temperature was observed at the middle of the canister side wall. The temperature of the buffer was lower than that of the canister, and the temperature increase of the deposition tunnel and the host rock was insignificant due to the lower effect of the heat source. The result of the thermal evolution of the EBS represented the highest thermal effects in the vicinity of the canister. In addition, the thermal effects were largely decreased after 10 years of the entire simulation period. It demonstrated that the model took 3 years to heat up the buffer around the canister. The temperature at the canister side wall increased until 3 years and then decreased after that time. This is because that the radioactive decay heat from the heat source was emitted enough to raise the overall temperature of the EBS by 3 years. However, the decay heat rate of the canister decreased exponentially with the disposal time and then its decay heat was not emitted enough after 3 years. In conclusion, the peak temperature results of the EBS were lower than 70°C to meet the technical design requirement.

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Keywords: Thermal analysis, Engineered barrier system, Deep geological repository, PFLOTRAN

Development of Performance Demonstration Laboratory for Engineered Barrier System in KAERI

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The engineered barrier system (EBS), composed of spent nuclear fuel, canister, buffer and backfill material, and near-field rock, plays a crucial role in the deep geological repository for high-level radioactive waste. Understanding the interactions between components in a thermo-hydro-mechanical-chemical (THMC) environment is necessary for ensuring the long-term performance of a disposal facility. Alongside the research project at KAERI, a comprehensive experimental facility has been established to elucidate the comprehensive performance of EBS components. The EBS performance demonstration laboratory, which installed in a 1,000 m², consists of nine experimental modules pertaining to rock mechanics, gas migration, THMC characteristics, buffer-rock interaction, buffer & backfill development, canister corrosion, canister welding, canister performance, and structure monitoring & diagnostics. This facility is still conducting research on the engineering properties and complex interactions of EBS components under coupled THMC condition. It is expected to serve as an important laboratory for the development of the key technologies for assessing the long-term stability of engineered barriers.

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Keywords: High-level nuclear waste, Deep geological repository, Engineered barrier system, Laboratory, Performance demonstration

Shear Characteristics of a Saw-cut Rock Discontinuity Depending on the Filling Material

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Rock discontinuities in underground rock behave as weak planes and affect the safety of underground structures, such as high-level radioactive waste disposal and underground research facilities. In particular, rock discontinuities can be a main flow path of groundwater and induce large deformation caused by stress disturbance or earthquakes. Therefore, it is essential to investigate the characteristics of rock discontinuities considering in-situ conditions when constructing high-level radioactive waste disposal, which needs to assure the long-term safety of the structure.

We prepared Hwang-Deung granite rock block specimens, including a saw-cut rock surface, to perform multi-stage direct shear tests as a preliminary study. In the multi-stage direct shear tests, we can exclude possible errors induced by different specimens for obtaining a full failure envelope by using an identical specimen. We applied the initial normal stress of 3 MPa on the specimen and increased the normal stress to 5 and 10 MPa step by step after peak shear stress observation. We obtained the mechanical properties of saw-cut rock surfaces from the experiments, including friction coefficient and cohesion. Additionally, we investigated the effect of filling material between rock discontinuities, assuming the erosion and piping phenomenon in the buffer material of the engineering barrier system. When the filling material existed in the rock surfaces, the shear characteristics deteriorated, and the effect of bentonite was dominant on the shear behavior.

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Keywords: Shear, Saw-cut, Discontinuity, Filling material

Suggestion on Design Alternative for Backfilling Solution of the Disposal Tunnel in the Deep Geological Repository for High-Level Radioactive Waste: Focusing on the Finnish Case

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In 2012, POSIVA selected a bentonite-based (montmorillonite) block/pellet as the backfilling solution for the deposition tunnel in the application for a construction license for the deep geological repository of high-level radioactive waste in Finland. However, in the license application (i.e. SC-OLA) for the operation submitted to the Finnish Government in 2021, the design for backfilling was changed to a granular mixture consisting of bentonite (smectite) pellets crushed to various sizes, based on NAGRA's buffer solution. In this study, as part of the preliminary design of the deep geological repository system in Korea, we reviewed history and its rationale for the design change of Finland's deposition tunnel backfilling solution.

After the construction license was granted by the Finnish Government in 2015, POSIVA conducted various lab- and full-scale in-situ tests to evaluate the producibility and performance of two design alternatives (i.e. block/pellet type and granular type) for backfilling. Principal demonstration tests and their results are summarized as follows: (a) Manufacturing of blocks using three types of materials (Friedland, IBeco RWC, and MX-80): Cracking and jointing under higher pressing loads were found. Despite adjusting the pressing process, similar phenomena were observed. (b) 1:6 scale experiment: Confirmation of density difference inhomogeneity due to the swelling of block/pellet backfill and void filling due to swelling behavior into the mass loss area of block/pellet. (c) FISST (Full-Scale In situ system Test): Identification of technical unfeasibility due to the inefficient (too manual) installation process of blocks/pellets and development of an efficient granular in-situ backfilling solution to resolve the disadvantage. (d) LUCOEX-FE (Large Underground Concept Experiments – Full-scale Emplacement) experiment: Confirmation of dense/homogeneous constructability and performance of granular backfilling solution.

In conclusion, the simplified granular backfill system is more feasible compared to the block/pellet system from the perspective of handling, production, installation, performance, and quality control. It is presumed that various experimental and engineering researches should be preceded reflecting specific disposal conditions even though these results are expected to be applied as key data and/or insights for selecting the backfilling solution in the domestic deep geological repository.

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Keywords: Deposition tunnel backfilling, Backfilling solution, Bentonite block, Bentonite pellet, Granular bentonite

Development of an Analytical Solution-Based Thermal Evolution Analysis Program for the Deep Geological Repository of High-Level Radioactive Waste in Korea

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With the importance of permanent disposal of high-level radioactive waste (HLW) generated in Korea, the deep geological disposal system based on the KBS-3 type is being developed. Since the deep geological repository must provide the long-term isolation of HLW from the surface environment and normal habitats for humans, plants, and animals, it is essential to assess the long-term performance of the disposal facility considering thermal-hydraulic-mechanical-chemical (T-H-M-C) evolution.

Decay heat dissipated from HLW contained in the canister causes an increase in temperature in the adjacent area. The requirement for the maximum temperature is established in consideration of the possibility of bentonite degradation. Therefore, when designing the repository, the temperature in the region of interest should be identified in detail through the thermal evolution assessment to ensure that the design requirement is satisfied. In the thermal evolution analysis, it is needed to evaluate the temperature distribution over the entire area of the disposal panel to consider the heat generated from both a single canister and adjacent canisters. Computational fluid dynamics (CFD) codes are widely used for detailed temperature analysis but are limited to simulating a wide range. Accordingly, in this study, we developed an analytical solution-based program for efficiently calculating the temperature distribution throughout the deposition panel, which is based on three-dimensional heat conduction equations.

The code developed can assess the temperature distribution of engineered and natural barrier systems. Principal parameters to be inputted are as follows: (a) geometry of the panel (e.g. width, length, height, spacing between canisters), (b) geometry of the canister (e.g. diameter, height), (c) thermal properties of bentonite and host-rock, (d) initial conditions (e.g. residual heat, temperature), and (e) time information (e.g. canister emplacement rate, time-interval, period). Through the calculation for the conceptual problem of a deposition panel capable of accommodating 900 (i.e. 30×30) canisters, it was confirmed that the program can adequately predict when and where the maximum temperature will occur.

It is expected that the overall temperature distribution within the panel can be obtained by the evaluation of the entire region using this program reflecting the detailed design of the repository to be developed in the future. In addition, the thermal evolution analysis considering the influence of other canisters can be performed by applying the results as boundary conditions in the CFD analysis.

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Keywords: HLW disposal, Deep geological repository, Thermal evolution analysis, Temperature distribution analysis program, 3-D heat conduction equation

Characterizing the Behavior of Buffer in Deep Geological Disposal Environment: Insight From THMC Interaction Experiment

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The buffer is installed around the disposal canister, subjected to heating due to decay heat while simultaneously experiencing expansion influenced by groundwater inflow from the surrounding rock. The engineering barrier system for deep geological disposal require the evaluation of long-term evolution based on the verification of individual component performance and the interactions among components within the disposal environment. Thus, it is crucial to identify the thermal-hydro-mechanical-chemical (THMC) processes of the buffer and assess its long- and short-term stability based on these interactions. Therefore, we conducted experimental evaluations of saturation-swelling, dry heating, gas transport, and mineralogical alterations that the buffer may undergo in the heated-hydration environment. We simulated a 310 mm-thick buffer material in a cylindrical form, simulating the domestic disposal system concept of KRS+ (the improved KAERI reference disposal system for spent nuclear fuel), and subjected it to the disposal environment using heating cartridges and a hydration system. To monitor the thermal-hydro-mechanical behavior within the buffer material, load cells were installed in the hydration section, and both of thermal couples and relative humidity sensors were placed at regular intervals from the heat source. After 140 days of heating and hydration, we dismantled the experimental cell and conducted post-mortem analyses of the samples. In this post-mortem analysis, we performed functions of distance from the water contents, heat source, wet density, dry density, saturation, and X-Ray diffraction analysis (XRD). The results showed that after 140 days in the heated-hydration environment, the samples exhibited a significant decrease water contents and saturation near the heat source, along with very low wet and dry densities. XRD Quantitative Analysis did not indicate mineralogical changes. The findings from this study are expected to be useful for input parameters and THMC interaction assessments for the long-term stability evaluation of buffer in deep geological disposal.

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Keywords: Disposal repository, Buffer, THMC, Post-mortem analysis

A Study on Enhancing Montmorillonite Grade Using Column Froth Flotation

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Montmorillonite, a versatile clay mineral with a wide range of industrial applications, is often found in natural deposits with impurities that limit its effectiveness. This study investigates the use of column froth flotation as an innovative technique to improve the purity of montmorillonite by selectively removing impurities without affecting its essential properties. Column froth flotation, a well-established mineral separation method, is adapted to address the specific challenges associated with enhancing montmorillonite purity. The process involves conditioning raw montmorillonite with carefully chosen reagents to selectively separate impurities, including quartz, feldspar, and other minerals commonly found alongside montmorillonite in natural deposits. Experimental results confirm the effectiveness of column froth flotation in significantly enhancing the purity of montmorillonite. This method allows for efficient impurity removal while preserving the essential properties of montmorillonite, making it suitable for various industrial applications. The study also explores the optimal conditions and reagent choices to maximize the purification process. In conclusion, column froth flotation offers a promising avenue for enhancing montmorillonite purity without compromising its fundamental properties. This study provides valuable insights into optimizing the process for large-scale industrial applications, facilitating the development of high-quality montmorillonite products tailored to specific industrial needs.

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Keywords: Montmorillonite, Column froth flotation, Mineral separation, Impurity removal, Purity enhancement

Characteristics of Fracture Apertures Under Various Normal Stress Conditions

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It is crucial to understand the hydro-mechanical behavior of rock mass to assess the performance of natural barriers. As rock fractures serve as both mechanically weak planes and prominent pathways for hydraulic flow, they significantly influence the hydro-mechanical behavior of the rock mass. Hence, understanding the characteristics of rock fractures is necessary to analyze the long-term behavior of natural barriers. In particular, fracture apertures are crucial parameters directly associated with groundwater flow and consequently hold significant importance in determining the hydro-mechanical behavior of natural barriers. Fracture apertures are defined as mechanical and hydraulic apertures, and various studies have been conducted to measure and analyze them. However, direct measurement of mechanical aperture according to changes in normal stress is known to be a challenging task. For this reason, there has been a scarcity of direct comparative findings between mechanical and hydraulic apertures under various normal stress conditions.

This study aims to analyze the characteristics of the mechanical and hydraulic apertures according to changes in normal stress based on experimental results. A digital analysis technique using a pressure film image was applied to analyze the mechanical aperture characteristics of the fracture. This technique can be applied by performing a pressure film compression test and a normal stiffness test on a fracture specimen, and has the advantage of being able to derive mechanical apertures under various normal stress conditions. The hydraulic aperture characteristics of the fracture were analyzed based on Cubic law after measuring the flow rate by performing a constant pressure injection test under triaxial compression conditions. By applying various confining pressures, it was possible to examine the hydraulic apertures according to changes in normal stress conditions. Through the experimental results, the relationship between the mechanical and hydraulic apertures of the fracture was summarized under various normal stress conditions. In addition, the experimental results were used to examine the applicability of various empirical equations for mechanical and hydraulic apertures proposed in previous studies.

The characteristics of the fracture aperture resulting from this study are significant because they are required in the hydro-mechanical model of natural barriers. Future studies will entail further experiments, with the objective of establishing novel relationships based on the accumulation of experimental data.

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Keywords: Natural barriers, Rock fractures, Mechanical aperture, Hydraulic aperture, Laboratory test

Developing a True Triaxial Experimental System and Research Strategy for Investigating Hydro-Mechanical Coupled Process in Natural Barrier

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The effectiveness of a crystalline natural barrier in providing sealing capabilities is based on the behavior of numerous fractures and their intersections within the rock mass. It is important to evaluate the evolving characteristics of fractured rock, as the hydro-mechanical coupled processes occurring through these fractures play a dominant role.

KAERI is actively developing a true tri-axial compression test system and concurrently conducting hydro-mechanical experiments using replicated fractured rock samples. This research is focused on a comprehensive examination of coupled processes within fractures, with a particular emphasis on the development of true tri-axial testing equipment. The designed test system has the capability to account for three-dimensional stress conditions, including vertical and both maximum and minimum horizontal principal stresses, realizing the disposal conditions at specific underground depths. Notably, the KAERI-designed test system employs the mixed true tri-axial concept, also known as the Mogi-type, which allows for fluid flow into fractures under tri-axial compression conditions. This system utilizes a hydraulic chamber to maintain constant stress in one direction through the application of oil pressure, while the other two directional stresses are applied using rigid platens with varying magnitudes. Once these mechanical stress conditions are established, control over fluid flow is achieved through the rigid platens in contact with the specimen section. This pioneering approach effectively replicates in-situ mechanical conditions while concurrently observing the internal fluid flow patterns within fractures, thereby enhancing our capacity to study these coupled phenomena.

As future research, numerical modeling efforts will be proceeding with experimental data-driven approaches to simulate the coupled behavior within the fractures. In these numerical studies, two distinct fracture geometry domains will be generated, one employing simplified rough-walled fractures and the other utilizing mismatched rough-walled fractures. These investigations mark the preliminary steps in the process of selecting and validating an appropriate numerical model for understanding the hydro-mechanical evolution within fractures.

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Keywords: True tri-axial test system, Fracture replica specimens, Hydro-mechanical evolution

Preliminary Study of Radiation/Heat Shielding Materials and Methods for Durable Monitoring Sensors for HLW Repository: A Material Approach

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In the nuclear environment, sensors ensure safety, monitoring, and operational efficiency under various operating conditions. These sensors come in various forms, each tailored to specific purposes, including nuclear safety and security, waste treatment and storage, gas leak detection, temperature and humidity monitoring, and corrosion detection. Ensuring the longevity of sensors without the need for frequent replacements is a vital goal for researchers in this field. This paper explores materials that can act as shields to protect sensors from harsh environmental conditions (high radiation and temperatures) to enhance their lifetime. The types of material that had been explored were divided into categories: metal and non-metal. Fourteen types of metal and seven different plastic materials were studied and focused on their characteristics and current applications. Considering properties like melting point, intensity, and conductivity, plastic materials are chosen to be examined as sensor shielding material.

A preliminary experiment was conducted to verify signal characteristics changes by shielding material. Metal material and plastic material each were placed in the middle of the granite and the target sensor. The result showed that when metal is between the granite and the sensor, the density and impedance are higher in granite than in the metal. This leads to signal attenuation and a shift in resonance frequency, while plastic does not. Therefore, PPS (Polyphenylene sulfide) and PAI (Polyamide-imide) have lower density and impedance than granite while also possessing heat, moisture, and radiation resistance for effective shielding.

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Keywords: Shielding material, Sensors, Impedance, Resonance frequency, PPS (polyphenylene sulfide), PAI (polyamide-imide)

Surrogate Modeling of Disposal System for Machine Learning Based Hybrid Solver Using U-Network

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Conducting a TSPA (Total System Performance Assessment) of the entire spent nuclear fuel disposal system, which includes thousands of disposal holes and their geological surroundings over many thousands of years, is a challenging task. Typically, the TSPA relies on significant efforts involving numerous parts and finite elements, making it computationally demanding. To streamline this process and enhance efficiency, our study introduces a surrogate model built upon the widely recognized U-network machine learning framework. This surrogate model serves as a bridge, correcting the results from a detailed numerical model with a large number of small-sized elements into a simplified one with fewer and large-sized elements. This approach will significantly cut down on computation time while preserving accuracy comparable to those achieved through the detailed numerical model.

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Keywords: TSPA, Disposal system, Machine learning, Hybrid solver, U-network

Suggestion on Minimizing Engineered Barrier System Degradation Based on Phenomenological Review for Copper Corrosion Mechanisms and Interaction of Resultant Product With Bentonite

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Copper, mainly used as a material for outer canister, generates various corrosion products under aerobic and anaerobic conditions in the operational and/or post-closure phases of the deep geological repository. These products could affect performance of engineering barrier system (EBS) through interaction with surrounding bentonite that makes up the buffer and backfill materials. Accordingly, in this study, we suggested research items to be conducted to minimize degradation of EBS due to copper corrosion products, based on the phenomenological review results for copper corrosion mechanisms and interaction between resultant product and bentonite in the deep geological disposal environment.

During the post-closure phase, condition in the disposal facility changes from aerobic to anaerobic over time, and thereby, causes and products of copper corrosion vary. Under aerobic condition, copper corrosion is mainly induced by oxygen (O_2) in the repository, chloride (Cl^-) and carbonate (CO_3^{2-}) ions from groundwater flowing into the facility, resulting in corrosion products such as cuprite (Cu_2O), tenorite (CuO), atacamite ($CuCl_2 \cdot 3Cu(OH)_2$) and malachite ($Cu_2CO_3(OH)_2$). And, copper corrosion under anaerobic condition is primarily due to hydrogen sulfide (H_2S) and sulfate (SO_4^{2-}) in groundwater flowing into the facility, leading to formation of chalcocite (Cu_2S) and covellite (CuS) as corrosion products.

Depending on environment of the disposal facility, copper corrosion products are dissolved and ionized to Cu^{2+} in groundwater, and subsequently adsorbed on the nearby smectite. Then, it causes a cation exchange reaction with exchangeable cations in the interlayer of smectite. As a result of reviewing the previous experiments, it was confirmed that Cu^{2+} -exchanged bentonite has a slightly reduced basal spacing and swelling capacity.

From the results as above, there is a possibility that performance of EBS may be degraded due to copper corrosion products. To minimize its effect of degradation in the domestic facility, items to be further studied are as follows: (a) Method for reducing copper corrosion such as selection of appropriate material and structure for the canister, and (b) How to control dissolution of copper canister product into groundwater through predicting type and ionization process.

The results of this study could be directly used to developing design concept of EBS for the domestic disposal facility and to establishing roadmap of future R&D programs.

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Keywords: Copper corrosion, Corrosion product, Bentonite degradation, Ion exchange

Methodology and Capability of APro's Dose Assessment Module

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The Korea Atomic Energy Research Institute (KAERI) is currently developing a process-based performance assessment model known as APro. Distinguished from the previous system-level safety assessment model developed by KAERI, APro exhibits the capacity to encompass a three-dimensional biosphere domain, evolving over the long term. In this study, we elucidate the methodology employed in developing the dose assessment module of APro and present the module's functionalities.

The procedural steps underlying radiation dose calculations within the APro framework can be succinctly outlined as follows: 1) Definition of a landscape model, utilizing information derived from a specified snapshot period provided by the APro biosphere transport module; 2) Generation of unit biotope objects spanning the landscape; 3) Evaluation of radionuclide transfer within the soil medium; 4) Calculation of activity concentration for flora and fauna groups; 5) Assessment of the distribution of effective dose among representative human groups; 6) Progressing through successive time steps.

The APro dose calculation module exhibits notable capabilities that encompass: 1) Accounting for radionuclide decay and ingrowth; 2) Facilitating transfer through unsaturated porous media; 3) Considering sorption effects; 4) Addressing the inheritance of radioactivity between various landscape models; 5) Offering customizable ecosystem parameters; 6) Providing flexibility for user-defined exposure pathways.

Leveraging these functionalities of the dose assessment module, APro is proficient in evaluating the distribution of radiological doses and associated risks for representative population groups, all while accounting for the dynamic, long-term evolution of the biosphere, including alterations in land cover.

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Keywords: Geological disposal system, Biosphere modeling, Radiological dose assessment, Performance assessment

Assessing the Geochemical Evolution of Engineered Barrier Systems Under Seawater Intrusion for High-Level Radioactive Waste Disposal Repository

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Understanding the long-term geochemical evolution of engineered barrier system is crucial for conducting safety assessment in high-level radioactive waste disposal repository. One critical scenario to consider is the intrusion of seawater into the engineered barrier system, which may occur due to global sea level rise. Seawater is characterized by its high ionic strength and abundant dissolved cations, including Na, K, and Mg. When seawater infiltrates an engineered barrier, such dissolved cations displace interlayer cations within the montmorillonite and affect to precipitation/dissolution of accessory minerals in bentonite buffer. These geochemical reactions change the porewater chemistry of bentonite buffer and influence the reactive transport of radionuclides when it leaked from the canister.

In this study, the adaptive process-based total system performance assessment framework (APro), developed by the Korea Atomic Energy Research Institute, was utilized to simulate the geochemical evolution of engineered barrier system resulting from seawater intrusion. Here, the APro simulated the geochemical evolution in bentonite porewater and mineral composition by considering various geochemical reactions such as mineral precipitation/dissolution, temperature, redox processes, cation exchange, and surface complexation mechanisms. The simulation results showed that the seawater intrusion led to the dissolution of gypsum and partial precipitation of calcite, dolomite, and siderite within the engineered barrier system. Additionally, the composition of interlayer cation in montmorillonite was changed, with an increase in Na, K, and Mg and a decrease in Ca, because the concentrations of Na, K, and Mg in seawater were 2-10 times higher than those in the initial bentonite porewater. Further studies will evaluate the geochemical sorption and transport of leaked uranium-238 and iodine-129 by applying TDB-based sorption model.

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Keywords: APro, Seawater intrusion, Geochemical evolution, Bentonite buffer, Safety assessment

Development of Geological Site Descriptive Model for Deep Geological Repository for Spent Nuclear Fuel

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The seven-year research project entitled “Development of workflow for integrated 3D geological site descriptive modeling” is being carried out from 2023. This research is funded by Ministry of Trade, Industry, and Energy (MOTIE). Progress of the research is discussed here.

The integrated 3D geological SDM (site descriptive model; GSDM hereafter) consists of three part; 1) three dimensional representation of geologic elements, 2) database for material properties and modeling results from SDMs of other disciplines (e.g., rock mechanics), and 3) a visualization tool for geology, material properties and modeling results. The GSDM is comparable to the GDSMs of SKB and POSIVA in its representation of geology by volume of geologic elements. However, our GSDM is different in that extra information of material properties and an extra tool for visualization is included in the GSDM. The rationale for incorporating material properties and a visualization tool into the GSDM is to expedite the development of the GSDM and SDMs of other disciplines by allowing single institution to integrate database and visualization with the GSDM.

SKUA-GOCAD is used for representation of geologic surfaces for ductile and brittle shear zones, and also for surfaces for delineation of volumes of rock units. We have adopted SKUA-GOCAD because the program offers powerful functions of interpolation including borehole data and geophysical prospecting. So far, we have tested the program for five different geologies, including sedimentary, high-grade metamorphic, and intrusive igneous geology. The test results are promising.

Incorporation of data and modeling results for the SDMs of other disciplines is at conceptual stage. The working conceptual model involves the following steps, 1) to provide the modeler of other disciplines with surface information representing geologic elements, 2) the modeler returns not only material properties but the results of numerical analysis, and 3) incorporation of material properties and modeling results into database. Since the numerical codes in other disciplines adopt different types of formats for 3D geology, we plan to adopt the widely used FEM format prepared by Gmsh. The visualization tool will also adopt Gmsh for graphical representation of 3D geology as well as database for material properties and modeling results.

When the working model of GSDM becomes available, rapid and significant progress is expected in the SDMs of other disciplines and related areas, for example, geotechnical investigation for deep geological repository.

Keywords: Geological site descriptive model, SKUA-GOCAD, Database, Visualization tool, Deep geological repository

Hydrogeochemical Background of a Tracer Test Site for Evaluating Buffering Capacity

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In the evaluation of the stability of radioactive waste disposal, it is imperative to take into account the concept of the redox front. Initially, this front is typically observed near the surface. However, if the hydraulic gradient increases due to the construction of a disposal facility, the redox front can potentially transport deeper into the geological environment through groundwater flow. This transport triggers changes in the geochemical characteristics, potentially diminishing the natural buffering capacity of the bedrock. Consequently, it is necessary to characterize both the unsaturated and saturated zones in the disposal site. In this context, a tracer test is a useful method to identify the characteristics of the site from the surface to the deep geological environment where the disposal facility can be located. Therefore, this study also aims to establish a methodology enabling a comprehensive understanding of the hydrogeochemical characteristics through the tracer test that can be applied to future sites for research URL (Underground Research Laboratory) or radioactive waste disposal in Korea.

For the tracer test, a UNIT (UNsaturated zone Insitu Test facility) was built within the KAERI and five wells with a depth of 24 m were installed in 2022. Before conducting the test, to determine the geochemical background characteristics of the site, topsoil and soils at depths of 30 cm, 60 cm, and 90 cm were collected. Additionally, a groundwater sample was obtained from the newly installed well. Soil samples were analyzed for soil texture, moisture content, total and exchange cations, anions, and heavy metals. Similarly, the groundwater sample was analyzed for cations, anions, and trace elements. The outcomes of these comprehensive analyses will serve as the baseline values in the hydrogeochemical changes after the tracer test. This includes changes in soil composition, water quality, precipitation/dissolution processes, and mineral phases. Furthermore, these results will be provided as input parameters for surface-underground interface models in future studies.

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Keywords: Hydrogeochemical characteristic, Soil, Groundwater, Reactive/non-reactive tracer test

Investigation of Radiation Effect on the HLW Repository Monitoring Sensors

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The high-level nuclear waste (HLW) repository disposes of high-level nuclear waste at a depth of 500 m to 1,000 m underground. Structural health monitoring must be accompanied by the complex environmental conditions of high temperature, high humidity, radiation, and mechanical stress. A thermocouple for measuring temperature, total stress meter and pore pressure meter for measuring stress and water pressure, relative hygrometer and electrical resistivity sensor (TDR or SUS) for measuring humidity, accelerometer for measuring crack signals, and strain gauge for measuring displacement are used.

For safety, after disposing of HLW in the HLW repository, access to the disposal tunnel gets blocked, making it impossible to replace or remove the monitoring sensors. So, it is necessary to evaluate the effect of the HLW repository's environmental conditions on the monitoring sensors and enhance their durability through quantitative life evaluation and shielding.

Before evaluating the life of accelerometers and strain gauges used in the HLW repository, an experimental study is conducted to determine failure modes and failure mechanisms under radiation conditions, which are unique environmental conditions of the HLW repository.

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Keywords: High-level nuclear waste repository, Monitoring sensor, Accelerated life test, Failure mode, Life prediction

Assessing the Impact of Host Rock Fracture Growth on Radionuclide Transportation From a High-Level Nuclear Waste Repository

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The increasing accumulation of spent nuclear fuel has raised interest in High-Level Waste (HLW) repositories. For example, Sweden is under construction of the KBS-3 repository. To ensure the safety of such HLW repository, various countries have been developing assessment models. In the Republic of Korea, the Korea Atomic Energy Research Institute has been developing on the A-KRS model. However, traditional safety assessment models have not considered the fracture growth in the far-field host rock as a function of time.

As repository safety assessments guarantee safety for million years, sustained stress naturally leads to the progressive growth of fractures as time goes on. Therefore, it becomes essential to account for fracture growth in the surrounding host rock. To address this, our study proposes a new coupling scheme between the Fracture growth model and the radionuclide transport model. That coupling scheme consists of the Cubic Law model as a fracture growth function and the GoldSim code which is a commercial software for radionuclide transport calculations. The model that adopting such fracture growth functions showed an increase of up to 15% in the release of radionuclide compared to traditional assessment models. our observations indicated that crack growth as a function of time led to an increase in hydraulic conductivity that allowed more radionuclide transport. Notably, these findings show the significance of adopting fracture growth models as a critical element in evaluating the safety of nuclear waste repositories.

Keywords: HLW, GoldSim, Host rock, Cubic law, Fracture growth

Correlation Between Uplift-Subsidence & Erosion-Deposition and Long-Term Evolution in Surface Environment Presented in IFEP List

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For the performance and safety assessments of deep geological disposal, developing scenarios, which represent possible long-term changes in the surface environment, is required. These scenarios are formulated using a list of FEPs (Features, Events, and Processes) that describes characteristics of disposal system components. In this study, using international FEP (IFEP) list from OECD/NEA, the individual FEPs related to uplift-subsidence and erosion-deposition were analyzed, and the correlation between each FEP was evaluated.

From the IFEP list, the elements related to uplift-subsidence and erosion-deposition processes that cause long-term changes in the surface environment were identified. Uplift-subsidence, erosion - deposition, and the long-term change factors caused by them were analyzed and a correlation diagram was produced according to their interactions. Basis for the integrated analysis of long-term changes in the surface environment and the construction of long-term change scenarios were established considering the evaluation of the factors that cause uplift-subsidence and erosion-deposition, and their correlation with the hydrology-hydrogeology, topography and local climate of the affected surface.

The results of this study will be used for systematically formulating scenarios of long-term changes in the surface environment due to uplift-subsidence and erosion-deposition based on natural phenomena. And, it may be necessary to modify and supplement the correlation of domestic FEPs based on the correlation diagram of IFEPs in order to analyze long-term changes in the surface environment in an integrated manner.

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Keywords: Deep geological disposal, Uplift-subsidence, Erosion-deposition, Surface environment, Long-term evolution, FEPs (features, events, processes)

Solid Phase Characterization and Groundwater Solubility of Synthetic Sklodowskite

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The disposal of spent nuclear fuel (SNF) in a deep geological repository (DGR) is a widely accepted strategy for the long-term sequestration of radiotoxic SNF. Ensuring the safety of a DGR requires the prediction of various reactions and migration behaviors of radionuclides (RNs) present in SNF within its geochemical surroundings. Understanding the dissolution behaviors of mineral phases harboring these RNs is crucial, as the levels of RNs in groundwater are basically linked to the solubility of these solid phases. Accurate measurements of solubility demand the use of well-defined solid materials characterized by chemical compositions and structures. Herein, we attempted the synthesis of sklodowskite, a magnesium-uranyl (U(VI))-silicate, employing a two-step hydrothermal synthetic approach documented previously. Subsequently, we subjected this synthesized sklodowskite to various analytical techniques, including powder X-ray diffraction (pXRD), scanning electron microscopy/energy dispersive X-ray spectrometry (SEM/EDX), and vibrational spectroscopies (FTIR and Raman). Based on our findings, we confidently identify the obtained mineral phase as sklodowskite ($\text{Mg}[\text{UO}_2\text{SiO}_3\text{OH}]_2 \cdot 5\text{H}_2\text{O}$). This identification is primarily based on the similarity between its pXRD pattern and the reference XRD pattern of sklodowskite. Furthermore, the measured infrared and Raman spectra show the vibrational modes of UO_2^{2+} and SiO_4^{4-} ions, particularly within the $700\sim 1,100\text{ cm}^{-1}$ region, which support that the synthetic mineral has a characteristic layered uranyl-silicate structure of crystalline sklodowskite. Finally, we utilized synthetic minerals to estimate its solubility up to about three months in a model groundwater, where the dissolved species composition is analogous to that of granitic groundwater from the KAERI Underground Research Tunnel. In this presentation, we will present in detail the results of spectroscopic characterizations and the methodology employed to assess the solubility of the U(VI)-silicate solid phase.

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Keywords: Magnesium uranyl silicate, Sklodowskite, X-ray diffraction, Solubility, FTIR/Raman spectroscopy, Time-resolved laser-induced luminescence spectroscopy

Temperature Effects on the Swelling Pressure of Compacted Bentonil-WRK Bentonite

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In environments where buffer materials are exposed to increased temperature due to the decay heat emitted by radioactive waste, it is crucial to assess the performance of the buffer material in relation to temperature effects. In this study, we conducted experiments using Bentonil-WRK, a calcium-type bentonite, compacted to a dry density of 1.65 g/cm^3 and an initial water content of 15%. The experimental temperature conditions were set to 30, 60, 90, 110, and 130°C . We observed that the swelling pressure of the compacted bentonite buffer decreased as the temperature increased. The findings from this study can provide valuable guidance for the design of high-level waste repository in Korea.

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Keywords: Bentonite, Swelling pressure, Buffer material, Deep disposal repository

Quantitative Evaluation of Structural Integrity in Nuclear Waste Disposal System Based on Adaptive AI to Improve Public Acceptance of Nuclear Safety

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The operation time of a disposal repository is generally more than one hundred years except for the institutional control phase. The structural integrity of a repository can be regarded as one of the most important research issues from the perspective of a long-term performance assessment, which is closely related to the public acceptance with regard to the nuclear safety. The objective of this study is to suggest the methodology for quantitative evaluation of structural integrity in a nuclear waste repository based on the adaptive artificial intelligence (AI), fractal theory, and acoustic emission (AE) monitoring. Here, adaptive AI means that the advanced AI model trained additionally based on the expert's decision, engineering & field scale tests, numerical studies etc. in addition to the lab. test.

In the process of a methodology development, AE source location, wave attenuation, the maximum AE energy and crack type classification were subsequently studied from the various lab. tests and Mazars damage model. The developed methodology for structural integrity was also applied to engineering scale concrete block (1.3 m × 1.3 m × 1.3 m) by artificial crack generation using a plate jacking method (up to 30 MPa) in KURT (KAERI Underground Research Tunnel). The concrete recipe used in engineering scale test was same as that of Gyeongju low & intermediate level waste repository. From this study, the reliability for AE crack source location, crack type classification, and damage assessment increased and all the processes for the technology development were verified from the Korea Testing Laboratory (KTL) in 2022.

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Keywords: Structural integrity, Adaptive AI, Waste disposal, Repository, Acoustic emission

A Case Study of Geological Long-Term Evolution Scenario of Foreign Countries and a Preliminary Application for KURT Site

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Advanced countries in the field of nuclear research and technology are currently examining the feasibility of deep geological disposal as the most appropriate method for the permanent management of high-level radioactive waste, with no intention of future retrieval. Deep geological disposal involves the placement of such waste deep underground within a stable geological formation, ensuring its permanent isolation from the human environment. To guarantee the enduring isolation and retardation of radionuclides with half-lives spanning tens of thousands to millions of years from the broader ecosystem, it is imperative to comprehend the long-term evolution of deep disposal systems, especially the role of natural barriers. These natural barriers, typically consisting of bedrock, encase the repository and undergo long-term evolutions due to tectonic movements and climate variations. For the effective disposal of high-level radioactive waste, a thorough assessment of the site's long-term geological stability is essential. This necessitates a comprehensive understanding of its tectonic evolution and development characteristics, including susceptibility to seismic and magmatic events like earthquakes and intrusions. Furthermore, a detailed analysis of alterations in the hydrogeological and geochemical environment resulting from tectonic movements over extended time frames is required to assess the potential for the migration of radionuclides.

In this paper, we have examined international evaluation methodologies employed to elucidate the predictive long-term evolution of natural barriers within disposal systems. We have extracted relevant methods from international case studies and applied a preliminary scenario illustrating the long-term evolution of the geological environment at the KURT (KAERI Underground Research Tunnel) site. Nevertheless, unlike international instances, the scarcity of quantitative data limits the depth of our interpretation. To present a dependable scenario in the future, it is imperative to develop predictive technologies aimed at comprehensively studying the geological evolution processes in the Korean peninsula, particularly within the context of radioactive waste disposal.

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Keywords: Natural barrier, Geological environment, Long-term evolution, Scenario, KURT

High-Resolution X-ray CT Scanning and Image Analysis for Qualitative Analysis of Internal Crack Patterns in Buffer Materials Following Gas Injection

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Engineered Barrier Systems (EBS) are a key element of deep geological repositories (DGR) and play an important role in safely isolating radioactive materials from the ecosystem. In the environment of a DGR, gases can be generated due to several factors, including canister corrosion. If the gas production rate exceeds the diffusion rate, pore pressures may increase, potentially inducing structural deterioration that impairs the function of the buffer material. Therefore, understanding the hydraulic-mechanical behavior of EBS due to gas generation is essential for evaluating the long-term stability of DGR.

This study employed X-ray computed tomography (CT) technology to observe cracks created inside the buffer material after laboratory-scale gas injection experiments. After CT scanning, we identified cracks more clearly using an image analysis method based on machine learning techniques, enabling us to examine internal crack patterns caused by gas injection.

In the samples observed in this study, no cracks were observed penetrating the entire buffer block, and it was confirmed that most cracks were created through the radial surface of the block. This is similar to the results observed in the LASGIT field experiment in which the paths of the gas migration were observed through the interface between the container and the buffer material.

This study confirmed the applicability of high-resolution X-ray CT imaging and image analysis techniques for qualitative analysis of internal crack patterns and cracks generated by gas breakthrough phenomena. This is expected to be used as basic data and crack analysis techniques in future research to understand gas migration in the buffer material.

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Keywords: Engineered barrier system, X-ray CT, Image analysis, Gas injection test, Crack propagation

Impact of Glacial Cycles and Sea Level Changes on Radionuclide Transport for Geological Disposal Systems

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Safety assessments for geological disposal systems extend over tens of thousands of years, taking into account the radiotoxicity decay period of spent nuclear fuel. During this extensive period, the biosphere experiences multiple glacial cycles, and fluctuations in seawater amounts, attributed to the formation and melting of glaciers, lead to global sea level changes known as eustasy. These sea level changes can directly influence the land-sea interface and groundwater flow dynamics, consequently affecting the pathways of radionuclide transport - an essential element of dose assessment. Therefore, this study aims to investigate how glacial cycles and sea level changes impact radionuclide transport within geological disposal systems, especially in the biosphere.

To achieve this objective, we obtained climate evolution data including sea level changes for the Korean Peninsula over a 200,000-years, simulated by a General Circulation Model (GCM). These data were then employed to predict site and hydrology evolutions. The study site was conceptualized biosphere of Artificial Disposal System (ADioS), and we utilized the Soil and Water Assessment Tool (SWAT) to simulate hydrological evolution. These datasets, encompassing climate, site, and hydrology evolution, were collectively employed as inputs for the biosphere module of Adaptive Process-Based Total System Performance Assessment Framework (APro). Subsequently, the APro's biosphere module calculated radionuclide transport in groundwater flow and its release into surface water bodies, considering the influences of glacial cycles and sea level changes.

The results show that hydrologic changes due to sea level change are relatively minor, while the impact of sea level change on groundwater flow and discharge is significant. Additionally, we identified that among the water bodies within ADioS, including rivers, lakes, and oceans, the ocean exhibits the most substantial radionuclide outflow throughout the entire period. The spatiotemporal distributions of radionuclides computed within APro will be further processed into a grid format and used as input for the dose assessment module.

Through this study, it was possible to determine the impact of long-term glacial cycles and sea level changes on radionuclide transport. Additionally, this module can serve as a valuable tool for providing the spatiotemporal variability of radionuclides required for enhanced dose assessments.

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Keywords: Biosphere, Glacial cycles, Sea level changes, Groundwater flow, Radionuclide transport, Safety assessment, Geological disposal system

Literature Review: A Development of Long-Term Evolution Scenarios in Biosphere Assessment

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The post-closure safety assessment of a repository is typically conducted over an extensive timescale from ten thousand to a million years. Considering that biosphere ecosystems may undergo significant changes over such lengthy periods, it is essential to incorporate the long-term evolution of the biosphere into the safety assessment. Climate change and landscape development are identified as critical drivers with the potential to impact the hydrogeological and hydrogeochemical characteristics of the biosphere. These changes can subsequently alter the migration patterns of radionuclides through the biosphere and influence human exposure doses. Therefore, this study formulates scenarios within the context of long-term biosphere evolution. We examine biosphere assessment processes employed in other countries and conduct a comparative study on scenario conditions. For example, biosphere assessment in Finland has identified sea-level changes and land-use alterations as significant factors in the long-term evolution of the biosphere. These factors are linked to Features, Events, and Processes (FEPs) associated with climate change and human activities. Sea-level changes are related to FEPs regarding climate change, land uplift, and shoreline displacement, while land-use changes are based on human activity-related FEPs (e.g., crop type, livestock and forest management, well construction, and demographics). Based on the literature review, this study has configured long-term evolution scenarios for the safety assessment of a deep geological repository for spent fuels.

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Keywords: Biosphere assessment, Scenarios, Climate change, Long-term evolution, Safety assessment

Determination of Critical Coagulation Concentration of WRK Bentonite Colloid

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In the high-level waste disposal systems, colloids generated through the chemical erosion of bentonite buffers can serve as critical mediators for the transport of radionuclides from the disposal environment to the biosphere. The stability of these colloids is influenced by the chemical composition of the groundwater. According to DLVO theory, the Critical Coagulation Concentration (CCC) is the ionic strength at which the total repulsive force between colloids is either less than or equal to the total attractive force. At ionic strengths lower than the CCC, electrostatic double-layer repulsion outweighs van der Waals attraction, forming a repulsive barrier between particles. Conversely, at ionic strengths higher than the CCC, attractive forces dominate, leading to particle aggregation. To investigate the CCC of bentonite colloids, this study focused on Ca-type WRK bentonite. Colloids separated from a ten g/L bentonite suspension underwent centrifugation (1,200 g for 30 minutes) and dialysis (3,500 MWCO) to produce colloid samples. After adjusting the ionic strength from 0.1 mM to 10 mM, the particle size distribution was monitored as a function of aggregation time for approximately 20 days. Rate constants, calculated based on variations in ionic strength, were used to interpret the observed results. The experimental outcomes revealed that the CCC value for WRK bentonite colloids was an order of magnitude lower with CaCl_2 than with NaCl. This suggests that Ca ions have a more significant impact on colloid stability, which has implications for the long-term safety of high-level waste disposal systems.

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Keywords: Colloid, Compacted bentonite, Chemical erosion, Ionic strength, Size distribution, Deep geological repository

Current Status of Colloid Formation and Migration International Joint Research Phase IV (2019–2023)

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The Colloid Formation and Migration (CFM) international joint research initiative continues as a part of the GTS's Radionuclide Retardation Programme, which has been in progress since 1984. This project focuses on examining the formation of colloids from a bentonite-engineered barrier system and exploring how these colloids impact the migration of radionuclides in fractured host rock when subjected to advective flow. Phase 1 of the project was launched in 2004 and concluded in early 2008, focusing on preliminary studies related to in-situ boundary conditions, predicting models, and supplementary lab works. Following that, Phase 2 spanned from 2008 to 2013 and aimed at fortifying the field setup by adding three new monitoring boreholes and suitable instrumentation in both the boreholes and tunnel. This phase also tested the system's resilience while mapping the flow domain. Phase 3 kicked off in January 2014 and extended until December 2018. During this period, the Long-term In-situ Test (LIT) was introduced in May 2014, featuring a set of compacted bentonite rings laced with radionuclide tracers. These were placed in a borehole to serve as a colloid and radionuclide source. CFM Phase 4 initiative commenced in January 2019, marking the successful deployment of the i-BET (In-situ Bentonite Erosion Test). This project component involves placing approximately 50 kg of compacted bentonite in a natural water-conducting shear zone. Korea Atomic Energy Research Institute (KAERI) joined CFM in 2008 to examine the behavior of colloid generation and migration with radionuclides in the Underground Research Laboratory. The fourth phase of the CFM project was also scheduled to include a post-mortem evaluation of the LIT and additional tracer experiments in the well-mapped MI shear zone. This study aims to provide an interim update on the ongoing i-BET, a key component of Phase 4 of the CFM project. We will also discuss the current status of the post-mortem analysis for the LIT experiment. In addition, we will outline plans for the forthcoming Phase VI of the project. These plans will continue to advance our understanding of radionuclide migration and the influence of bentonite-based disposal systems.

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Keywords: Colloid, Migration, Radionuclides, Bentonite, Long-term in-situ test

Morphological Characteristics of Tensile Fractures Based on X-ray Imaging

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The occurrence of shear failure in a rock mass, resulting from the sliding of joint surfaces, is primarily influenced by the surface roughness and contact area of these joints. Furthermore, since joints serve as crucial conduits for the movement of water, oil, gas, and thermal energy, the aperture and geometric complexity of these joints have a significant impact on the hydraulic properties of the rock mass. This renders them critical factors in related industries. Therefore, to gain insights into the mechanical and hydraulic behavior of a rock mass, it is essential to identify the key morphological characteristics of the joints mentioned above. In this study, we quantified the morphological characteristics of tensile fractures in granitic rocks using X-ray CT imaging. To accomplish this, we prepared a cylindrical sample of Hwang-Deung granite and conducted splitting tests to artificially create tensile fractures that closely resemble rough joint surfaces. Subsequently, we obtained 2D sliced X-ray CT images of the fractured sample with a pixel resolution of approximately 0.06 mm. By analyzing the differences in CT numbers of the rock components (e.g., fractures, voids, and rock matrix), we isolated and reconstructed the geometric information of the tensile fracture in three dimensions. Finally, we derived morphological characteristics, including surface roughness, contact area, aperture, and fracture volume, from the reconstructed fracture.

Acknowledgements

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Keywords: Tensile fracture, Granitic rock, Fracture morphology, X-ray imaging, 3D reconstruction, Contact area

Laboratory Tests for Determining Temperature Limits of Fiber Optic Accelerometer for HLW Disposal Repository

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High level radioactive waste (HLW) final disposal repository is faced thermos-hydro-mechanical - radioactive condition because it is placed over 500 m in depth and waste emits decay heats for decades. Repository will be operated around 100 years and will be closed after all the wastes are disposed. The integrity of engineered barriers including buffer, backfill, concrete plug and canister and natural barrier (natural rock mass) will be stood during operating periods. Monitoring sensors for concrete and rock mass is conducted using piezo based sensors such as accelerometer or acoustic emission (AE) sensors. Typical accelerometer for harsh conditions is commonly expensive and data/power cable can be a potential groundwater inflow and nuclide outflow path. The fiber optic accelerometer whose data and power cable are united and has limited volume. Therefore, it can be a potential alternative sensor of piezo based sensors. The temperature limits and accelerated tests for fiber optic sensors are conducted. Most of sensors gives a malfunction around 130°C. The results of these experimental tests give a possibility of communications in compacted bentonite buffer and will be utilized for the design of monitoring systems for the repository.

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Keywords: HLW, Repository, Accelerometer, Fiber optic, Temperature

Numerical Model Development for Long-Term Erosion of Bentonite Buffer

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Effective containment and disposal of high-level radioactive waste is critical to ensure long-term environmental and human safety. Especially bentonite, which is widely used as a buffer material due to its favorable characteristics such as swelling ability and low permeability, plays an important role in preventing the migration of radioactive waste into the surrounding environment. However, the long-term performance of bentonite buffer remains an area of ongoing investigation, with particular attention focused on erosion mechanisms induced by swelling and groundwater flow. The erosion of the bentonite buffer can significantly impact the integrity of buffer and lead to the formation of colloids, which could potentially facilitate the transport of radionuclides through groundwater. Therefore, quantification of bentonite buffer erosion based on an understanding of the underlying mechanisms and factors that influence bentonite buffer erosion, is essential for the safety assessment of high-level radioactive waste repositories.

In this study, we aimed to develop a bentonite buffer erosion model using the Adaptive Process-based total system performance assessment framework for a geological disposal system (APro) proposed by the Korea Atomic Energy Research Institute (KAERI). The impact of bentonite erosion on performance assessment can be broadly divided into bentonite property degradation by the penetration of the bentonite buffer into rock fractures and the formation of pseudocolloids. To simulate this phenomenon, Two-region model based on a dynamic bentonite diffusion model is adopted, which can quantify the extent of bentonite intrusion and loss by erosion. Using this Two-region model, a numerical model was developed to simulate the degradation of bentonite properties based on the amount of bentonite intrusion, as well as to simulate the migration of pseudocolloids in the near-field by deriving the amount of pseudocolloid production based on the loss of bentonite and the sorption rate of radionuclides. To check the applicability of the developed numerical model, preliminary analysis was performed for the effect of bentonite erosion in terms of process-based performance assessment. It is anticipated that this comprehensive model developed in this study will contribute to the accurate and reliable assessment of the long-term performance and safety of high-level radioactive waste repositories.

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Keywords: High-level radioactive waste repository, Process-based total system performance assessment, APro, Bentonite buffer erosion, Two-region model

Utilization of Coupled Reactive Transport Model for the Safety Assessment of High-Level Radioactive Waste Repository

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The effect of various physicochemical processes, such as seawater intrusion, on the performance of the engineered barrier should be closely analyzed to precisely assess the safety of high-level radioactive waste repository. In order to evaluate the impact of such processes on the performance of the engineered barrier, a thermal-hydrological-chemical model was developed by using COMSOL Multiphysics and PHREEQC. The coupling of two software was achieved through the application of a sequential non-iterative approach. Model verification was executed through a comparative analysis between the outcomes derived from the developed model and those obtained in prior investigations. Two data were in a good agreement, demonstrating the model is capable of simulating aqueous speciation, adsorption, precipitation, and dissolution. Using the developed model, the geochemical evolution of bentonite buffer under a general condition was simulated as a base case. The model domain consists of 0.5 m of bentonite and 49.5 m of granite. The uraninite (UO₂) was assigned at the canister-bentonite interface as the potential source of uranium. Assuming the lifetime of canister as 1,000 years, the porewater mixing without uranium leakage was simulated for 1,000 years. After then, the uranium leakage through the dissolution of uraninite was initiated and simulated for additional 1,000 years. In the base case model, where the porewater mixing between the bentonite and granite was the only considered process, the gypsum tended to dissolve throughout the bentonite, while it precipitated in the vicinity of bentonite-granite boundary. However, the precipitation and dissolution of gypsum only showed a limited effect on the performance of the bentonite. Due to the low solubility of uraninite in the reduced environment, only infinitesimal amounts of uranium dissolved and transported through the bentonite. Additional cases considering various environmental processes, such as seawater or cement porewater intrusion, will be further investigated.

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Keywords: High-level radioactive waste repository, Bentonite, Reactive transport model, Radionuclide

Hydration and Dehydration of Montmorillonite With Saline Fluids Under P-T-X Conditions Relevant to the Deep Geological Repository Environment

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Bentonite, primarily composed of montmorillonite, plays a crucial role as one of the engineering barrier materials in a deep geological repository (DGR). The advantageous properties of montmorillonite, such as its swelling capacity, low permeability, and low thermal conductivity, make it a key component as a buffer material for isolating high-level radioactive waste from the surrounding natural environment. It has been known that the stability of montmorillonite is influenced by a wide range of pressure-temperature-composition (P-T-X) conditions relevant to the DGR environment. When considering potential geological events, of notable concerns are its interactions with groundwater or seawater at elevated temperatures, leading to safety hazards within the system. In this study, therefore, we investigated the hydration and dehydration reactions of Ca-montmorillonite with saline fluids such as NaCl and KCl solutions at elevated pressures and temperatures by conducting in-situ X-ray diffraction experiments using both a capillary sample heating cell and a resistive-heated diamond anvil cell. As a result, we observed different hydration states of montmorillonite depending on the chemical composition of fluids, i.e., tri-hydrated layers in NaCl and bi-hydrated layers in KCl solutions, respectively. Furthermore, we identified that montmorillonite undergoes distinct stepwise dehydration with increasing temperature, and the dehydration temperature of montmorillonite significantly increases with increasing water pressure. Consequently, this study would provide insights into the stability of hydrated montmorillonite in a seawater-dominated fluid environment and its implications for the long-term safety of the disposal system.

Keywords: Bentonite, Montmorillonite, Hydration, Dehydration

Preliminary Application of the Brittle Structure Boundary Definition Method Using Boreholes Around the KAERI Underground Research Tunnel

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The radwaste repository consists of a multi-barrier, including natural and engineered barriers. The repository's long-term safety is ensured by using the isolation and delay functions of the multi-barrier. Among them, natural barriers are difficult to artificially improve and have a long time scale. Therefore, in order to evaluate its performance, site characteristics should be investigated for a sufficient period using various analytical methods. Natural barriers are classified into lithological and structural characteristics and investigated. Structural factors such as fractures, faults, and joints are very important in a natural barrier because they can serve as a flow path for groundwater in performance evaluation. Considering the condition that the radioactive waste repository should be located in the deep part, the drill core is an important subject that can identify deep geological properties that could not be confirmed near the surface.

However, in many previous studies, a unified method has not been used to define the boundaries of structural factors. Therefore, it is necessary to derive a method suitable for site characteristics by applying and comparing the boundary definition criteria of various structural factors to boreholes. This study utilized the 1,000 m deep AH-3 and DB-2 boreholes and the 500 m deep AH-1 and YS-1 boreholes drilled around the KURT (KAERI Underground Research Tunnel) site. Methods applied to define the brittle structure boundary include comparing background levels of fracture and fracture density, excluding sections outside the zone of influence of deformation, and confining the zone to areas of concentrated deformation. All of these methods are analyzed along scanlines from the brittle structure. Deriving a site-specific method will contribute to reducing the uncertainties that may arise when analyzing the long-term evolution of brittle structures within natural barriers.

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Keywords: Natural barrier, Long-term evolution, Brittle structure, Brittle deformation, Borehole

Radionuclide Transport Modeling in Excavation-Damaged Zone

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Deep disposal facility for High-Level radioactive Waste (HLW) uses a multi-barrier system to prevent the leakage of radionuclide. As a part of the engineered barrier, bentonite is primarily considered as the main buffering material. This is due to the adsorption and swelling properties of the bentonite, which are expected to effectively impede leakage of the radionuclide. In many cases, adsorption is generally regarded as occurring only within the buffer zone. However, several research has been conducted to explore the possibility of bentonite intrusion into the Excavation-Damaged Zone (EDZ) generated during excavation processes, because of the swelling properties of the bentonite. Generally, for host rock near the deep disposal facility such as granite, groundwater flows through the fracture network. Therefore, analysis of the characteristics of the fracture network is essential for predicting the behavior of radionuclide in groundwater. Accordingly, the bentonite intrusion into the fracture network is critical for safety assessment of the deep disposal facility. To analyze this, hydro-geochemical model was established utilizing COMSOL Multiphysics and PHREEQC, observing changes of the behavior of U (VI) along fracture network due to the swelling of bentonite. Modeling was conducted with progressively changing intrusion depth of the bentonite.

According to the results, the behavior of U (VI) exhibited significant changes depending on the connectivity of the fractures. Based on the distribution characteristics of the fracture network, heterogeneous groundwater flow was observed. U (VI) was transported through the preferential pathway, which indicates high connectivity, due to the rapid groundwater flow. Notably, when changing the intrusion depth of bentonite, significant differences in behavior of U (VI) were observed in the 0-20 cm case. In contrast, as the intrusion depth increased, it was observed that differences became less evident. These results indicate that changes in the properties of fracture network in EDZ due to the swelling of bentonite significantly influence the behavior of U (VI).

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Keywords: High-level radioactive waste, EDZ, Uranium, Reactive transport, COMSOL Multiphysics, PHREEQC

Suggestions on the Performance Demonstration Test of Backfill Materials in the Deep Geological Repository for High-Level Radioactive Waste in Korea

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Properties of bentonite, mainly used as buffer and/or backfill materials, will evolve with time due to thermo-hydro-mechanical-chemical (THMC) processes, which could deteriorate the long-term integrity of the engineered barrier system. In particular, degradation of the backfill in the evolution processes makes it impossible to sufficiently perform the safety functions assigned to prevent groundwater infiltration and retard radionuclide transport. To phenomenologically understand the performance degradation to be caused by evolution, it is essential to conduct the demonstration test for backfill material under the deep geological disposal environment. Accordingly, in this paper, we suggest types of tests and items to be measured for identifying the performance evolution of backfill for the Deep Geological Repository (DGR) in Korea, based on the review results on the performance assessment methodology conducted for the operating license application in Finland.

Some of insights derived from reviewing the Finnish case are as follows: 1) The THMC evolution characteristics of backfill material are mainly originated from hydro-mechanical and/or hydro-chemical processes driven by the groundwater behavior. 2) These evolutions could occur immediately upon installation of backfill materials and vary depending on characteristics of backfill and groundwater. 3) Through the demonstration experiments with various scales, the hydro-mechanical evolution (e.g. advection and mechanical erosion) of the backfill due to changes in hydraulic behavior could be identified. 4) The hydro-chemical evolution (e.g. alteration and microbial activity) could be identified by analyzing the fully-saturated backfill after completing the experiment.

Given the findings, it is judged that the following studies should be first conducted for the candidate backfill materials of the domestic DGR. a) Lab-scale experiment: Measurement for dry density and swelling pressure due to saturation of various backfill materials, time required to reach full saturation, and change in hydraulic conductivity with injection pressure. b) Pilot-scale experiment: Measurement for the mass loss due to erosion; Investigation on the fracture (piping channel) forming and resealing in the saturation process; Identification of the hydro-mechanical evolution with the test scale. c) Post-experiment dismantling analysis for saturated backfill: Measurement of dry density, and contents of organic and harmful substances; Investigation of water content distribution and homogenization of density differences; Identification of the hydro-chemical evolution with groundwater conditions.

The results of this study could be directly used to establishing the experimental plan for verifying performance of backfill materials of DGR in Korea, provided that the domestic data such as facility design and site characteristics (including information on groundwater) are acquired.

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Keywords: Backfill materials, Long-term evolution, Performance assessment, Demonstration test, Deep geological repository

Analysis of Legal Frameworks and Site Development Systems in Foreign Countries for High-Level Radioactive Waste Disposal Site Acquisition

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For the successful establishment of deep geological repositories for High-Level Radioactive Waste (HLW), a foundational step involves instituting a sequenced site development strategy that complies with the standards detailed in the corresponding legal systems.

This study reviews the International Atomic Energy Agency (IAEA)'s suggested framework for site development and dissects the legal architecture and staged procedures prevalent in front-runner nations—Sweden, Finland, and Switzerland—in the realm of HLW disposal. Furthermore, based on the analysis, the distinct legal standards and site selection processes for each country have been examined.

These global cases reveal each nation has developed and implemented laws, methods, and technologies tailored to their specific circumstances, geological conditions, and requirements. Notably, Sweden, Finland, and Switzerland emphasize the essence of transparent, open dialogues with local communities and residents. Such transparency is pivotal in ensuring the communities and residents grasp disposal safety nuances, thereby overcoming the barriers of community endorsement in final site selection.

Insights from comparing the site selection cases in these leading countries are anticipated to provide invaluable data and information for devising an optimized site development system tailored to Korea's conditions and realities.

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Keywords: High level radioactive waste, Deep geological repositories, Legal standards, Site selection processes

Management of Input Uncertainty in Performance Assessment of Deep Geological Repository

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The deep geological repository for high-level radioactive waste requires careful consideration due to its exceptionally long-term implications, making long-term impact assessments essential. However, evaluating the long-term effects of deep geological repositories using performance assessment models is accompanied by various sources of uncertainty, including uncertainties about the future, model uncertainties, and uncertainties associated with input data. These multifaceted uncertainties arise from factors such as a lack of current knowledge, contributing to a complex web of unpredictability. Managing, mitigating, and ultimately eliminating these uncertainties is crucial for ensuring the performance and safety of deep geological repositories. Currently, the Korea Radioactive Waste Agency (KORAD) is developing a complex behavior model that incorporates Thermal-Hydraulic-Mechanical-Chemical (THMC) phenomena within the disposal system using PFLOTRAN. To address model uncertainties and furthermore input data uncertainties for this intricate model, an automated sensitivity analysis system has been developed. This automated system operates without human intervention, facilitating tasks such as automatic parameter adjustments and the quantification of uncertainties. Furthermore, this system aids in identifying key factors characterized by substantial uncertainties. Through this system, it is possible to examine concentration distributions in each components of the deep disposal facility in response to changes in input data and to identify factors with significant uncertainties. The sensitivity results and key uncertainty factors obtained through this system are intended to be used for optimizing uncertainties in future research and development.

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Keywords: Deep geological disposal, Uncertainty, Performance assessment, Sensitivity analysis, Quantification of uncertainty, PFLOTRAN

A Methodology Using Machine Learning for System-Level Safety Assessment Considering Complex Processes in the Near-Field of Geological Disposal System

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Over the years, in the field of safety assessment of geological disposal system, system-level models have been widely employed, primarily due to considerations of computational efficiency and convenience. However, system-level models have their limitations when it comes to phenomenologically simulating the complex processes occurring within disposal systems, particularly when attempting to account for the coupled processes in the near-field. Therefore, this study investigates a machine learning-based methodology for incorporating phenomenological insights into system-level safety assessment models without compromising computational efficiency. The machine learning application targeted the calculation of waste degradation rates and the estimation of radionuclide flux around the deposition holes. To develop machine learning models for both degradation rates and radionuclide flux, key influencing factors or input parameters need to be identified. Subsequently, process models capable of computing degradation rates and radionuclide flux will be established. To facilitate the generation of machine learning data encompassing a wide range of input parameter combinations, Latin-hypercube sampling will be applied. Based on the predefined scenarios and input parameters, the machine learning models will generate time-series data for the degradation rates and radionuclide flux. The time-series data can subsequently be applied to the system-level safety assessment model as a time table format. The methodology presented in this study is expected to contribute to the enhancement of system-level safety assessment models when applied.

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Keywords: Safety assessment model, Machine learning, Complex processes, Near-field, Geological disposal system

Evaluation of Unconfined Compression Test for Bentonite Mixtures

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The bentonite buffer material is a crucial component for disposing of high-level radioactive waste (HLW). Several additives have been proposed to enhance the performance of bentonite buffer materials. In this study, unconfined compression tests were conducted on bentonite mixtures as well as pure bentonite buffer material. Joomunjin and silica sands were added at a 30% ratio, and graphite was added at 3% along with bentonite. The unconfined compression strength (UCS) and elastic modulus of pure bentonite were found to be 20% to 50% higher than those of bentonite mixtures under similar dry density and water content conditions. This decrease in strength can be attributed to the reduced cross-sectional area available for bearing the applied load in the bentonite-mixture. Furthermore, the 3% graphite-bentonite mixture exhibited a 10% to 30% higher UCS and elastic modulus compared to the 30% sand-bentonite mixtures. However, since the strength properties of additive-bentonite mixtures are lower than those of pure bentonite, it is essential to evaluate thermohydraulic-mechanical functional criteria when considering the use of bentonite mixtures as buffer materials.

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Keywords: Bentonite mixtures, Unconfined compression test, Unconfined compression strength

Numerical Modelling of Nuclide Transport Considering Long-Term Evolution in Far-Field of Disposal System

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The safety of deep geological disposal systems has to be ensured to guarantee the isolation of radionuclides from human and related environments for over a million years. Over such a long timeframe, disposal systems can be influenced by climate change, leading to significant long-term impacts on the hydrogeological condition, including changes in temperature, precipitation and sea levels. These changes can affect groundwater flow, alter geochemical conditions, and directly/indirectly impact the stability of the repository. Hence, it is essential to conduct a safety assessment that considers the long-term evolution induced by climate change.

In this context, the Korea Atomic Energy Research Institute (KAERI) is developing the Adaptive Process-based total system performance assessment framework for a geological disposal system (APro). Currently, numerical modules for APro are under development to account for the long-term evolution that can influence groundwater flow and radionuclide transport in the far-field of the disposal system. This study focuses on the development of two numerical modules designed to model permafrost formation and buoyancy force due to relative density changes.

Permafrost is defined as a ground in which temperature remains below zero-isotherm (0°C) continuously for more than two consecutive years. In regions where permafrost forms, the relative permeability of porous media is significantly reduced. The changes in permeability due to permafrost formation are modelled by calculating the unfrozen fluid content within a porous medium. Meanwhile, buoyancy force can occur when there is a difference in density at the boundary of two distinct water groups, such as seawater (salt water) and freshwater. Sea level change associated with climate change can alter the boundary between seawater and freshwater, resulting in changes in groundwater flow. The buoyancy force due to relative density is modelled by adjusting concentration boundary conditions.

Using the developed numerical modules, we evaluated the long-term evolution's effects by analyzing radionuclide transport in the far-field of the disposal system. Incorporating permafrost and buoyancy force modelling into the APro framework will contribute valuable insights into the complex interactions between geological and climatic factors, enhancing our ability to ensure the secure isolation of radionuclides for extended periods.

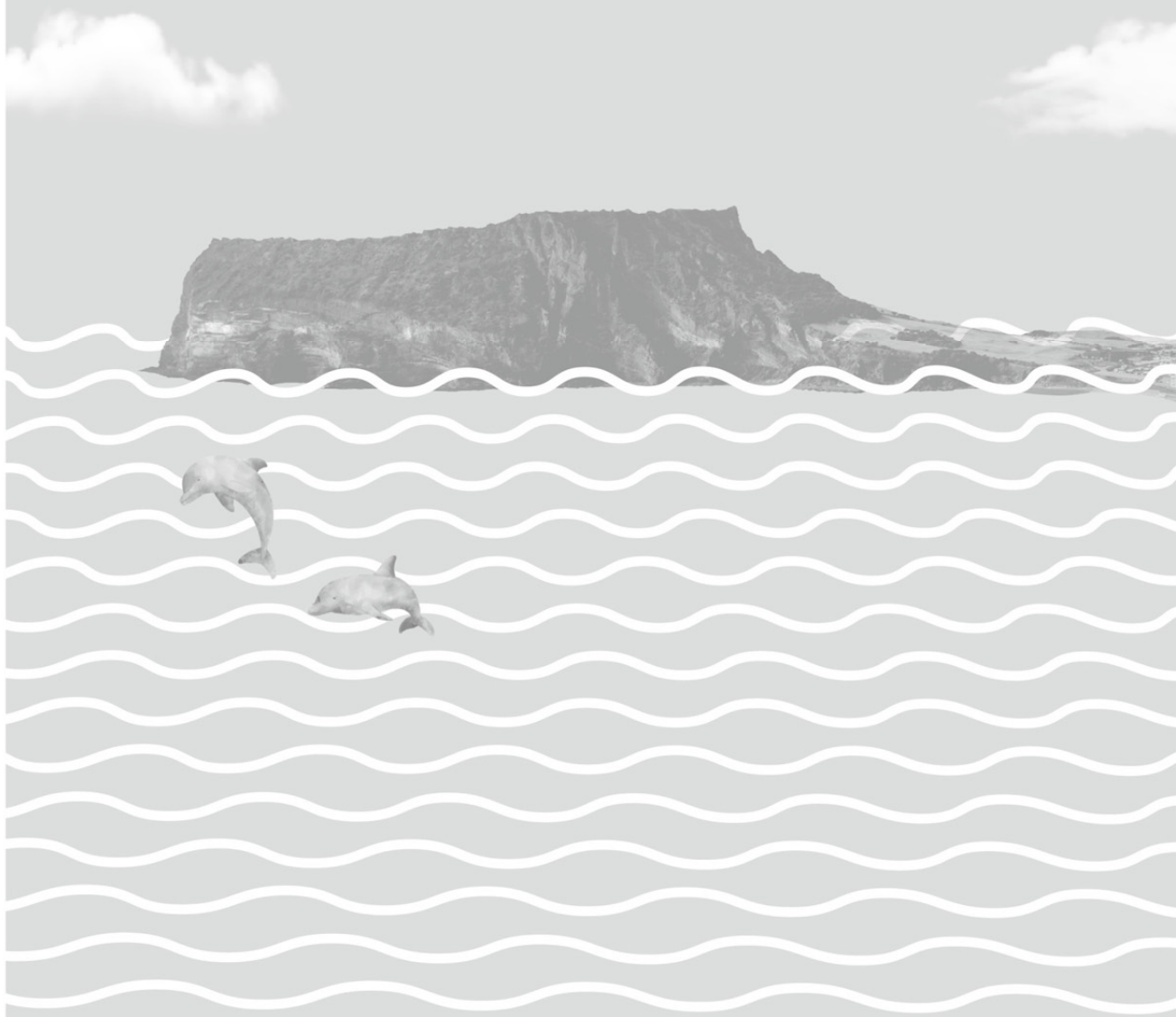
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Keywords: APro, Long-term evolution, Permafrost, Buoyancy force, Nuclide transport

4분과

중저준위폐기물관리 (ORAL)



Development and Performance Test of an Equipment for a Selective Separation of Paraffin From the Paraffin Solidification Product Using the Vacuum Evaporation and Condensational Deposition Recovery Method

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Nuclear power plants in Korea stores approximately 3,800 drums of paraffin solidification products. Due to the lack of homogeneity, these solidification products are not allowed to be disposed of. There is therefore a need for the separation of paraffin from the solidification products. This work developed an equipment for a selective separation of paraffin from the solidification product using the vacuum evaporation and condensational recovery method in a closed system. The equipment mainly consists of a vacuum evaporator and a condensational deposition recovery chamber. Non-isothermal vacuum TGAs, kinetic analyses and kinetic predictions were conducted to set appropriate operation conditions. Its basic operability under the established conditions was first confirmed using pure paraffin solid. Simulated paraffin solidification product fixing dried boric acid waste including nonradioactive Co and Cs were then fabricated and tested for the capability of selective separation of paraffin from the simulated waste. Paraffin was selectively separated without entertainment of Co and Cs. It was confirmed that the developed equipment could separate and recover paraffin in the form of nonradioactive waste.

Keywords: Paraffin solidification product, Paraffin, Vacuum evaporation, Kinetic analysis, Kinetic prediction, Closed system

A Background Study of VLLW Spent Resin Solidification

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Domestic waste acceptance criteria (WAC) require flowable or homogeneous wastes, such as spent resin, concentrated waste, and sludge, etc., to be solidified regardless of radiation level, to provide structural integrity to prevent collapse of repository, and prevent leaching. Therefore, very-low level (VLL) spent resin (SR) would also require to be solidified. However, such disposal would be too conservative, considering IAEA standards do not require robust containment and shielding of VLL wastes. To prevent unnecessary cost and exposure to workers, current WAC advisable to be amended, thus this paper aims to provide modified regulation based on reviewed engineering background of solidification requirement.

According to NRC report, SR is classified as wet-solid waste, which is defined as a solid waste produced from liquid system, thus containing free-liquid within the waste. NRC requires liquid wastes to be solidified regardless of radiation level to prevent free liquid from being disposed, which could cause rapid release of radionuclides. Furthermore, considering class A waste does not require structural integrity, unlike class B and C wastes, dewatering would be an enough measure for solidification. This is supported by the cases of Palo Verde and Diablo Canyon nuclear power plants, whose wet-solid wastes, such as concentrated wastes and sludge, are disposed by packaging into steel boxes after dewatering or incineration. Therefore, dewatering VLL spent resin and packaging them into structural secure packaging could satisfy solidification goal.

Another goal of solidification is to provide structural support, which was considered to prevent collapse of soil covers in landfills or trenches. However, providing structural support via solidification agent (ex. Cement) would be unnecessary in domestic 2nd phase repository. As the domestic 2nd phase repository is cementitious structure, which is backfilled with cement upon closure, the repository itself already has enough structural integrity to prevent collapse.

Goldsim simulation was run to evaluate radiation impact by VLL SR, with and without solidification, by modelling solidified wastes with simple leaching, and unsolidified wastes with instant release. Both simulations showed negligible impact on radiation exposure, meaning that solidifying VLL SR to delay leaching would be irrational.

Therefore, dewatering VLL SR and packaging it into a secure drum (ex. Steel drum) could achieve solidification goals described in NRC reports and provide enough safety to be disposed into domestic repositories. In future, the studied backgrounds in this paper should be considered to modify current WAC to achieve efficient waste management.

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Keywords: Spent resin, Solidification, WAC, Radioactive waste, Safety assessment

K-CRAFT: A Toolbox for Safety Assessment of Radioactive Waste Processing Complex Based on IAEA's SADRWMS Methodology

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Among nuclear power plants in the Republic of Korea, Kori Unit 1 and Wolsong Unit 1 have been permanently shut down, and Kori Unit 1 is preparing to be decommissioned. According to the decommissioning plan (DP) of Kori Unit 1, a radioactive waste processing complex will be built on the Kori site to reduce radioactive waste generated during decommissioning actively, and various types of decommissioning waste are expected to be treated in the complex. It is judged that matters related to the safety assessment of the complex are not included in the DP since the equipment and treatment processes have not been determined.

IAEA GSR Part 5 states that radioactive waste processing complex shall be operated according to national regulations and the conditions imposed by the regulatory body. However, it has been confirmed that separate regulatory requirements for the complex have not yet been established in Korea. It is expected that the Regulation on Technical Standards for Nuclear Facilities, etc. will be applied mutatis mutandis. Liquid and gaseous radioactive materials can be expected to be released into the sea or atmosphere during the operation of the complex. Accordingly, it should be proved that standards such as discharge limits of radioactive effluents are met. Although the assessment of radioactive effluent discharged from nuclear power plants to the environment is systematically conducted, it has been confirmed that the safety assessment framework for radioactive effluents discharged from the complex has not yet been established.

Currently, the SAFRAN Tool is based on SADRWMS (Safety Assessment Driving Radioactive Waste Management Solutions), an IAEA safety assessment methodology for pre-disposal management, which uses Pathway Dose Factors (PDFs) derived from generic environmental models. Therefore, in order to conduct a more detailed safety assessment of the complex on a specific site, site characteristic data should be reflected. Although safety assessment using the SAFRAN Tool was conducted at the Thailand Institute of Nuclear Technology (TINT) facility, detailed data were not provided, and PDFs reflecting site characteristic data were not applied. Also, no other studies that considered many types of waste and provided detailed data on the safety assessment were not confirmed.

Therefore, this study developed K-CRAFT (Kyung Hee – Comprehensive RAdioactive waste treatment Facility safety assessment Tool), this tool that can derive PDFs by reflecting site characteristic data based on the SADRWMS methodology and conducted preliminary safety assessment for the complex which will be built in Kori site by this tool.

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Keywords: SADRWMS, Radioactive waste processing complex, K-CRAFT, Pathway dose factors, Decommissioning

Review of Requirement for Radiation Stability Test of Solidified Wastes Form in the Silo Type Disposal Facility Waste Acceptance Criteria for LILW

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Structural stability of a waste form can be provided by the waste form itself (steel components, etc.), by processing the waste to a stable form (solidification, etc.), or by emplacing the waste in a container or structure that provides stability (HICs or engineered structure, etc.). The waste or container should be resistant to degradation caused by radiation effects.

In accordance with the requirements for the domestic waste acceptance criteria, irradiation testing of solidified waste forms containing spent resin should be conducted on specimens exposed to a dose of $1.0\text{E}+6$ Gy and other material $1.0\text{E}+7$ Gy.

Expected cumulative dose over 300 years is about $1.770\text{E}+6$ Gy for spent resin and $0.770\text{E}+6$ Gy for dried concentrated waste generated from NPPs generally.

According to NRC Waste Form Technical Position, to ensure that spent resins will not undergo adverse degradation effects from radiation, resins should not be generated having loadings that will produce greater than $1\text{E}+6$ Gy total accumulated dose. If it necessary to load resins higher than $1\text{E}+6$ Gy, it should be demonstrated that the resin will not undergo radiation degradation at the proposed higher loading. This is the recommended maximum activity level for organic resins based on evidence that while a measurable amount of damage to the resin will occur at $1\text{E}+6$ Gy, the amount of damage will have negligible effect on disposal site safety. Cementitious materials are not affected by gamma radiation to in excess of $1\text{E}+6$ Gy. Therefore, for cement-stabilized waste forms, irradiation qualification testing need not be conducted unless the waste forms contain spent resins or other organic media or the expected cumulative dose on waste forms containing other materials is greater than $1\text{E}+7$ Gy. Testing should be performed on specimens exposed to $1\text{E}+6$ Gy or the expected maximum dose greater than $1\text{E}+6$ Gy for waste forms that contain ion exchange resins or other organic media or the expected maximum dose greater than $1\text{E}+7$ Gy for other waste forms.

This is suggestion as a review result that requirement for irradiation testing of solidified waste forms has something to be revise in detail and definitively.

Keywords: Solidified waste form, Irradiation test, Waste acceptance criteria, LILW, Disposal

Disposal Assessment of β Radionuclide-Removed Waste From Spent Resin Tank at a Wolsong Nuclear Power Plant

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We conducted safety assessments for the disposal of spent resin mixed waste after the removal of beta radionuclides (^3H , ^{14}C) in a landfill facility. The spent resin tank of Wolsong nuclear power plant is generated by 8:1:1 weight ratio of spent ion exchange resin, spent activated carbon and zeolite.

Waste in the spent resin tank was classified as intermediate-level radioactive waste due to ^{14}C . Other nuclides such as ^{60}Co and ^{137}Cs exhibit below the low-level radioactive waste criteria. The techniques for separating mixed waste and capturing ^{14}C have been under development, with a particular focus on microwave-based methods to remove beta radionuclides (^3H , ^{14}C) from spent activated carbon and spent resin within the mixed waste. The spent resin and activated carbon within the waste mixture exhibits microwave reactivity, heated when exposed to microwaves. This technology serves as a means to remove beta isotopes within the spent resin, particularly by eliminating ^{14}C , allowing it to meet the low-level radioactive waste criteria. Using this method, the waste mixture can meet disposal requirements through free water and ^3H removal.

These assessments considered the human intrusion scenarios and were carried out using the RESRAD-ONSITE code. The institutional management period after facility closure is set at 300 years, during which accidental exposures resulting from human intrusion into the disposal site are accounted for.

The assessment of radiation exposure to intruders in a landfill facility included six human intrusion scenarios, such as the drilling scenario, road construction scenario, post-drilling scenario, and post-construction scenario.

Among the six human intrusion scenarios considered, the most conservative assessment about annual radiation exposure was the post-drilling scenario. In this scenario, human intrusion occurs, followed by drilling and residence on the site after the institutional management period. We assumed that some of the vegetables and fruits grown in the area may originate from contaminated regions. Importantly, we confirmed that radiation doses resulting from post-institutional management period human intrusion scenarios remain below 0.1 mSv/y, thus complying with the annual dose limits for the public. This research underscores the importance of effectively managing and securing radioactive waste, with a specific focus on the safety of beta radionuclide-removed waste during long-term disposal, even in the face of potential human intrusion scenarios beyond the institutional management period.

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This work was supported by the Korea Institute of Energy Technology Evaluation and Planning and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (grant no. 202300 233879).

Keywords: Disposal assessment, Spent resin tank, Human intruder scenario, RESRAD-ONSITE

Optimization of Metakaolin-Based Geopolymer for Solidification of Cesium and Molybdenum

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The sustainability of the nuclear power industry hinges increasingly on the safe, long-term disposal of radioactive waste. Despite significant innovations and advancements in nuclear fuel and reactor design, the quest for a permanent solution to handle accumulating radioactive waste has received comparatively less attention. Conventionally, two widely recognized solidification methods, namely cementation for low and intermediate-level waste and vitrification for high-level waste, have been favored due to their simplicity, affordability, and availability.

Recently, geopolymers have emerged as an appealing alternative, gaining attention for their minimal carbon footprint, robust chemical and mechanical properties, cost-effectiveness, and capacity to immobilize a broad spectrum of radionuclides, including radioactive organic compounds. This study delves into the synthesis of metakaolin-based geopolymers tailored for the immobilization of fission products like cesium (Cs) and molybdenum (Mo). The investigation unfolded in two key steps.

In the initial step, we optimized the alkali content to prevent the occurrence of efflorescence, a potential issue. Remarkably, as the $\text{Na}_2\text{O}/\text{Al}_2\text{O}_3$ ratio increased from 0.82 to 1.54, we observed significant enhancements in both compressive strength (11.45 to 27.07 MPa) and density (up to 2.23 g/cm^3). This suggests the importance of careful adjustment in achieving the desired geopolymer characteristics.

The second phase involved the incorporation of 2wt% of Cs and Mo, both individually and as a mixture, into the geopolymer matrix. We prepared the GP paste, which was poured into cylindrical molds and cured at 60°C for one week. To scrutinize the crystallinity, phase purity, and bonding type of the developed matrix, we employed XRD and FTIR techniques. Additionally, we conducted standard compressive strength tests (following ASTM C39/C39M-17b) to assess the stacking durability and robustness of the developed waste form, vital for storage, handling, transportation, and disposal in a deep geological repository.

Furthermore, to evaluate the chemical durability, diffusivity and leaching of the GP waste matrix, we employed the ASTM standard Product Consistency Test (PCT: C 1285-02) and American nuclear society's devised leaching test (ANS 16.1). It is noteworthy that the introduction of Cs and Cs/Mo in the GP matrix led to a reduction of more than 50% and 60% in compressive strength, respectively. This outcome may be attributed to the interference of Cs and Mo with the geopolymerization process, potentially causing the formation of new phases. However, it is crucial to emphasize that both developed matrices exhibited an acceptable normalized leaching rate of less than $10^{-5} \text{ g}\cdot\text{m}^{-2}\cdot\text{d}^{-1}$. This finding underscores the promising potential of the GP matrix for the immobilization of cationic and anionic radioactive species, paving the way for more sustainable nuclear waste management practices.

Acknowledgements

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Keywords: Fission products, Geopolymers, Solidification, Radioactive waste management

Resin Withdrawal From Storage Tanks Using SRHS (Spent Resin Handling System) and High Purity CO₂ Capture in PHWR

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Nuclear power plants use ion exchange resins to purify liquid radioactive waste generated while operating nuclear power plants. In the case of PHWR, ion exchange resins are used in heavy water and dehydration systems, liquid waste treatment systems, and heavy water washing systems, and the used ion exchange resins are stored in waste resin storage tanks. The C-14 radioactivity concentration in the waste resin currently stored at the Wolsong Nuclear Power Plant is 4.6×10^6 Bq/g, exceeding the low-level limit, and if all is disposed of, it is 1.48×10^{15} Bq, exceeding the total limit of 3.04×10^{14} Bq of C-14 in the first stage disposal facility. Therefore, disposal is not possible at domestic low/medium-level disposal facilities. In addition, since the heavy water reactor waste resin mixture is stored at a ratio of about 20% activated carbon and zeolite mixture and about 80% waste resin, mixture extraction and separation technology and C-14 desorption and adsorption technology are required.

Accordingly, research and development has been conducted domestically on methods to treat heavy water waste resin, but the waste resin mixture separation method is complex and inefficient, and there are limitations in applying it to the field due to the scale of the equipment being large compared to the field work space. Therefore, we would like to introduce a resin treatment technology that complements the problems of previous research.

Previously, the waste resin mixture was extracted from the upper manhole and inspection hole of the storage tank, but in order to improve limitations such as worker safety, cost, and increased work time, the SRHS, which was planned at the time of nuclear power plant design, is utilized. In addition, by capturing high-purity ¹⁴CO₂ in a liquid state in a high-pressure container, it ensures safety for long-term storage and is easy to handle when necessary, maximizing management efficiency.

In addition, the modularization of the waste resin separation and withdrawal process from the storage tank, C-14 desorption and monitoring process, high-concentration ¹⁴CO₂ capture and storage process, and ¹⁴CO₂ adsorption process enables separation of each process, making it applicable to narrow work spaces.

When this technology is used to treat waste resin mixtures in PHWR, it is expected to demonstrate its value as customized, high-efficiency equipment that can secure field applicability and safety and reflect the diverse needs of consumers according to changes in the working environment.

Keywords: PHWR, Spent resin, Carbon-14, Disposal safety, Recycle

Immobilization of Radioactive Iodine With Copper Tellurite Glass

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Radioactive iodine-129, a byproduct of nuclear fission in nuclear power plants, presents significant environmental and health risks due to its high solubility in water and volatility. Iodine-129, with its half-life of 1.57×10^{17} years, necessitates safe management and disposal. Therefore, safely capturing and managing I-129 during spent nuclear fuel reprocessing is of paramount importance. To address these challenges, various glass waste forms containing silver iodide have been developed, such as borosilicate, silver phosphate, silver vanadate, and silver tellurite glasses. These glasses effectively immobilize iodine, but the high cost of silver raises affordability concerns. This study introduces $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glass waste forms for iodine immobilization, a novel approach. The cost-effectiveness of copper, in contrast to silver, makes it an attractive alternative. The $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glass waste forms were synthesized with varying CuI content (x) in $(1-x)(0.3\text{Cu}_2\text{O} \cdot 0.7\text{TeO}_2)$ glass matrices. X-ray diffraction (XRD) confirmed amorphous structures, and X-ray fluorescence (XRF) quantified composition. X-ray photoelectron spectroscopy (XPS) and Raman spectroscopy provided insights into structural properties. Durability assessments using a 7-day product consistency test (PCT-A) and inductively coupled plasma-mass spectrometry (ICP-MS) revealed compliance with U.S. glass regulations, making $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glasses a promising choice for iodine immobilization in radioactive waste.

Keywords: Radioactive iodine-129, Copper iodide, Tellurite glass, Glass durability test

Microwave Processing for Recovery of C-14 From Spent Ion Exchange Resin

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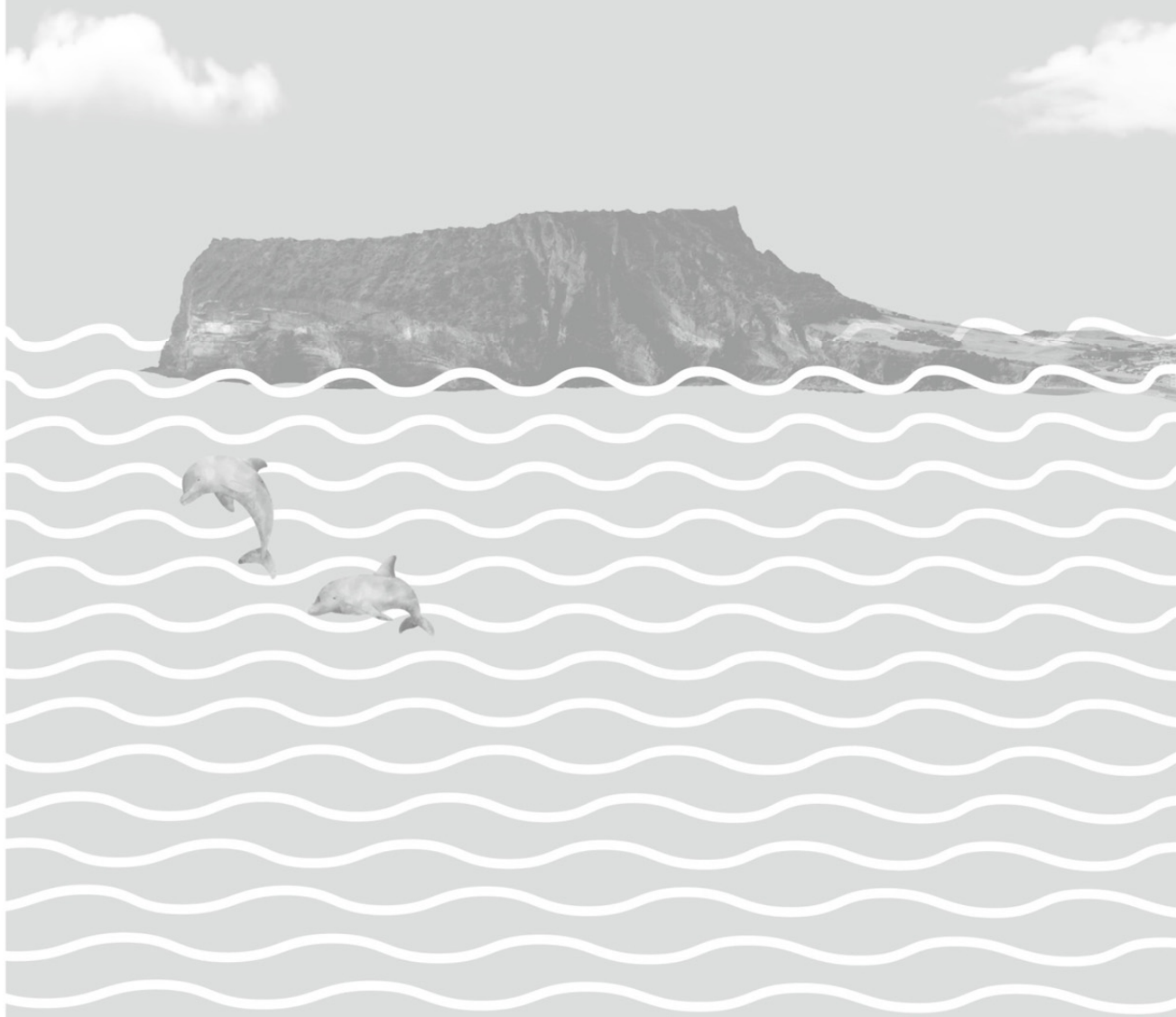
Ion exchange resins are commonly employed in the treatment of liquid radioactive waste generated in nuclear power plants (NPP). The ion exchange resin used in NPP is a mixed-bed ion exchange resin known as IRN-150, which is of nuclear grade. This resin is a mixture of cation exchange resin and anion exchange resin. The cation exchange resin removes cationic radionuclides such as Cs and Co, while anion exchange resin handles anions (e.g., $\text{H}^{14}\text{CO}_3^-$), effectively purifying the liquid waste. Spent ion exchange resins (spent resin) containing C-14 are classified as low and intermediate level radioactive waste, and their radioactivity needs to be reduced as it exceeds the disposal limit regulated by law. Therefore, the microwave technology for the removal of C-14 from spent resin has been investigated. Previous studies have successfully developed a method for the effective removal of C-14 during the resin treatment process. However, it was observed that, in this process, functional groups in the resin were also removed, resulting in the generation of off-gases containing trimethylamine. These off-gases can dissolve in water from process, increasing its pH, which can subsequently hinder the recovery of C-14.

In this study, we investigated the high-purity recovery of C-14 by adjusting the moisture content within the reactor following microwave treatment. Mock spent resins, consisting of 100 g of resin with HCO_3^- ion-exchanged and 0, 25, or 50 g of deionized water, were subjected to microwave treatment for 40 or 60 minutes. Subsequently, the C-14 desorption efficiency of the mock spent resins was evaluated using an acid stripping process with H_3PO_4 solution. The functional group status of the mock spent resins was analyzed using ^{15}N NMR spectroscopy. The results showed that the mock spent resins exhibited efficient C-14 recovery without significant functional group degradation. The highest C-14 desorption efficiency was achieved when 25 g of deionized water was used during microwave treatment.

Keywords: Spent ion-exchange resin, C-14, Microwave, Intermediated level radioactive waste

4분과

중저준위폐기물관리 (POSTER)



Development of a Zero Liquid Discharge System for Treating Wastewater Originating From Uranium Contaminated Soil

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The primary objective of this study is to evaluate a systematic design's effectivity in remediating actual uranium-contaminated soil. The emphasis was placed on practical and engineering aspects, particularly in assessing the capabilities of a zero liquid discharge system in treating wastewater derived from soil washing. The research method involved a purification procedure for both the uranium-contaminated soil and its accompanying wastewater. Notably, the experimental outcomes demonstrated successful uranium separation from the contaminated soil. The treated soil could be self-disposed of, as its uranium concentration fell below $1.0 \text{ Bq} \cdot \text{g}^{-1}$, a level endorsed by the International Atomic Energy Agency for radionuclide clearance. The zero liquid discharge system's significance lay in its distillation process, which not only facilitated the reuse of water from the separated filtrate but also allowed for the self-disposal of high-purity Na_2SO_4 within the residues of the distilled filtrate. Through a comparative economic analysis involving direct disposal and the application of a remediation process for uranium-contaminated soil, the comprehensive zero liquid discharge system emerged as a practical and viable choice. The successful demonstration of the design and practicality of the proposed zero liquid discharge system for treating wastewater originating from real uranium-contaminated soil is poised to have a lasting impact.

Acknowledgements

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Keywords: Zero liquid discharge, Soil washing, Uranium precipitation, Wastewater treatment

Development Status of Off-Site Dose Calculation Program

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In order to evaluate the exposure dose of residents living near nuclear power plants, a Off-site Dose Calculation Program (ODCP) has been developed based on SAP since 2021. The ODCP consists of social environmental factor, atmospheric diffusion factors, liquid/gas dose evaluation, and comprehensive analysis, and was developed by dividing it into functional modules. The off-site dose calculation can be carried out monthly, quarterly, semi-annual, and annual, and resident dose evaluation is conducted by entering air diffusion factors and emissions for each period. It also enables comprehensive evaluation result management by developing history management functions together.

Keywords: Off-site dose calculation program, Nuclear power plant

Evaluation of Separation/Concentration Treatment System to 10 m³ Scale Boron-Containing Nuclear Power Plant Discharge Water

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Domestic nuclear power plants can affect the environment if multiple devices are operated on one site and even a trace amount of pollutants that may affect the environment after power generation are simultaneously discharged. Therefore, not only radioactive substances but also ionic substances such as boron should be discharged as minimally as possible.

We adopted pilot CDI and SD-ELIX system to separating and concentrating of boron containing nuclear power plant discharge water.

The boron concentration of the initial inflow water tended to decrease over time. The water quality of concentrated water also reached its peak until the initial 60 minutes, but tended to decrease in line with the decrease in the inflow water concentration.

The boron removal rate was in the range of 85 to 99% with respect to the initial boron concentration of 15 to 25 mg/L. On the other hand, performance degradation due to the use of electrochemical modules is also observed, and regeneration through low ion-containing water cleaning effective.

We shortened processing time by considering the optimal flow rate conditions and conductivity conditions and converting electrochemical modules into series or parallel.

Acknowledgements

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Keywords: Boron, Separation/concentration, 10 m³, NPP discharge water

Development of a Conceptual Design for the Solidification Facility of Long-Stored Radioactive Waste

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The development of existing radioactive waste (RI waste) management technologies has been limited to processing techniques for volume reduction. However, this approach has limitations as it does not address issues that compromise the safety of RI waste management, such as the leakage of radioactive liquid, radiation exposure, fire hazards, and off-gas generation. RI waste comes in various forms of radioactive contamination levels, and the sources of waste generation are not fixed, making it challenging to apply conventional decommissioning and disposal techniques from nuclear power plants. This necessitates the development of new disposal facilities suitable for domestic use.

Various methods have been considered for the solidification of RI waste, including cement solidification, paraffin solidification, and polymer solidification. Among these, the polymer solidification method is currently regarded as the most suitable material for RI waste immobilization, aiming to overcome the limitations of cement and paraffin solidification methods. Therefore, in this study, a conceptual design for a solidification system using polymer solidification was developed. Taking into account industrial applicability and process costs, a solidification system using epoxy resin was designed.

The developed solidification system consists of a pre-treatment system (fine crush), solidification system, cladding system, and packing system. Each process is automated to enhance safety by minimizing user exposure to radioactive waste. The cladding system was designed to minimize defects in the solidified material. Based on the proposed conceptual design in this paper, we plan to proceed with the specific design phase and manufacture performance testing equipment based on the basic design.

Acknowledgements

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Keywords: Sealed radioactive source, Polymer solidification, Waste treatment, RI animal carcass, RI liquid organic waste

Process for Regulatory Clearance of Radioactive Waste at KAERI

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There is a large amount of radioactive waste in waste storage in the Korea Atomic Energy Research Institute. Some of the radioactive waste was generated during the dismantling process due to Korea Research Reactor 1&2 and it accounts for 20% of the total waste. Radioactive waste must be reduced by appropriate disposal methods to secure storage space and to reduce disposal costs. Research Reactor wastes include wastes that are below the acceptable criteria for self-disposal and non-contaminated wastes, so they can be treated as wastes subject to self-disposal through contamination analysis and reclassification. In order to deregulation radioactive waste, it is necessary to meet the self-disposal standards stipulated in the Domestic Nuclear Act and the treatment standards of the Waste Management Act. The main factors of deregulation are surface contaminant, radionuclide activity and dose assessment. To confirm the contamination of waste, surface contaminant and gamma nuclide analysis were performed. After homogenizing the waste sample, it was placed in 1 L Mariinelli beaker. When collecting waste samples, 1 kg per 200 kg of waste was collected. The concentrations of the major radionuclides Co-60, Cs-134, Cs-137, Eu-152, and Eu-154 were analyzed using HPGe detector. To evaluate radiation dose, various computational programs were used. A dose assessment was performed with the analyzed nuclide concentration. The concentrations of representative nuclides satisfied the deregulation acceptance criteria and the results of the dose assessment corresponding to self-disposal method was also satisfied. Based on this results, KAERI submitted the report on waste self-disposal plan to obtain approval. After final approval, Research Reactor waste is to be incinerated and incineration ash is to be buried in the designated place. Some metallic waste has been recycled. In this study, the suitability of deregulation for self-disposal was confirmed through the evaluation of the surface contaminant analysis, radionuclide concentration analysis and dose assessment.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 521210-23).

Keywords: Wood waste, Clearance, Self-disposal

Assessment of Structural Improvements in Disposal Containers Through Metal Composite Coating

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Radioactive waste is typically disposed of using standard 200 and 320 L drums based on acceptance criteria. However, there have been no cases evaluating the disposal and suitability of 200 L steel drums for RI waste disposal. There has been a lack of prior assessments regarding the disposal and suitability of 200 L steel drums for the disposal of RI waste. Radioactive waste is transported to disposal facilities after disposal in containers, where the drums are loaded and temporarily stored. Subsequently, after repackaging the disposal drums, the repackaged drums are transported to disposal facilities by vehicle or ship for permanent disposal.

Disposal containers can be susceptible to damage due to impacts during transportation, handling, and loading, leading to potential damage to the radiation primer coating during loading. Additionally, disposal containers may be subject to damage from electrochemical corrosion, necessitating the enhancement of corrosion resistance. Metal composite coatings can be employed to enhance both abrasion resistance and corrosion resistance. The application of metal composite coatings to disposal containers can improve the durability and radiation shielding performance of radioactive waste disposal containers.

The thickness of radioactive waste disposal containers is determined through radioactive shielding analysis during the design process. The designed disposal containers undergo structural analysis, considering loading conditions based on the disposal environment. This paper focuses on evaluating the structural improvements achieved through the implementation of metal composite coatings with the goal of enhancing corrosion and abrasion resistance.

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Keywords: Sealed radioactive source, Waste treatment, RI waste, Disposal container

Calculation of Radioactive Sensitivity According to Irradiated Stainless Steel-304 Impurity Isotopes

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The decommissioning of nuclear power plants will generate a lot of low and intermediate-level radioactive waste (LILW), and preliminary radioactive evaluation for these wastes should be carried out before decommissioning work. Mainly, Concrete, Carbon Steel, Stainless Steel-304 (SS304) and Inconel are used in many parts of nuclear power plants and considered as main resource of nuclear wastes. Depending on the material location, the number of neutrons irradiated to material varies, which can range from self-disposal waste to LILW.

In this paper, activation analysis was performed to compare the radiation dose according to the presence or absence of impurity elements present in SS304. For the calculation, SS304 composition and impurity elements were used as described in the report of NUREG-3474. This report lists 41 impurity elements for SS304 and other materials. Calculation code is used ORIGEN-S module in SCALE 6.1 code. Neutron flux is used as arbitrary value that around $1\text{E}+11$ level and irradiation time is set as 30 year with 10-year cooling time. In the ORIGEN-S calculation, 1g of SS304 is used for easy calculation of specific activity.

The ORIGEN-S calculation results are as follows. All impurity elements contained case calculated $9.32\text{E}+07$ Bq activity. In the absence of all impurity elements case and most cases shows that total Becquerel value after 10-year cooling time around $9.11\text{E}+07$ Bq, and Co impurity case had larger result. The calculation was performed again by increasing the amount of impurity substances by 100 times to perform the sensitivity evaluation more reliably. Representatively, Li, N, Co, and Ba impurity elements cases were calculated to have a particularly high Becquerel. Especially Co impurity element case, a total Becquerel of $3.03\text{E}+08$ was calculated. Accordingly, evaluation of impurities mixed in SS304 must be considered, and in particular, the inclusion rate for Co must be considered.

Acknowledgements

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Keywords: Activation analysis, SCALE, SS304, Impurity

Review on the Latch Housing Treatment Among Radioactive Waste Generated During Replacement Reactor Vessel Closure Head (RRVCH)

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Among the nuclear power plant facility improvement projects, out of a total 10 replacement reactor vessel closure head (RRVCH), five have been replaced, starting with Gori Unit 1, and five, including Hanul Unit 1, Hanbit Units 5 and 6, and Hanul Units 3 and 4 will be replaced in the future. This paper presents the method of treating Latch Housing among radioactive waste generated during the replacement of Hanul Unit 2 (February 2023). Latch Housing controls the control rod by receiving magnetic force from the CRDM's Coil Stack. Located in the Old Reactor Vessel Head (ORVH) Hot Spot, the range of measured radiation dose rate was 0.3 to 0.8 mSv h⁻¹ (up to 4.5 mSv h⁻¹). The amount of radioactive waste generated was 35.8 Baler-Drum (based on 200L), and the order of treatment was to cut into the Omega Seal of the CRDM, the CRDM and Latch Housing Transfer, the boundary of the CRDM and Latch Housing, the Rod Travel Housing, the Motor Housing and the Latch Assembly, and then transfer and Drumming. In the United States, out of 93 operating reactors, 31 reactor vessel heads have been replaced and 19 reactor vessel heads are scheduled to be replaced. In Korea, 25 reactors are in operation, and two reactors have been permanently shut down. Among them, the nine old reactors for more than 30 years (as of September 2021) are expected to achieve ALARA and reduce radwaste management costs through the management method applied to replace the reactor vessel head.

Keywords: Nuclear power plant, Reactor vessel head, CRDM, Latch assembly, Radioactive waste

Effect of Concentration of Cesium, Strontium, and Cobalt Inside the Cement Solidified Waste on the Leachability Index

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Low- and intermediate level waste (LILW) repository in Gyeongju, Korea is in operation and the radioactive waste should satisfy the waste acceptance criteria (WAC) of the repository. Among the WAC of the Gyeongju LILW repository, the leachability index criterion is considered to be the criterion that is directly related to the isolation of the radionuclides from biosphere. Cesium, strontium, and cobalt should satisfy the leachability index larger than six by following the ANS 16.1 leaching test method. Several research were performed for the leachability index of Cs, Sr, and Co by following the ANS 16.1 leaching test method. However, the test condition of the previous research is expected to be different to the condition of the actual waste.

Due to the radioactivity of the radionuclide such as Cs, Sr and Co, most of the research applied the surrogate of those radionuclides. The concentration of those nuclides was generally measured by the inductively coupled plasma (ICP) equipment, however, high concentration compared to the disposal limit of those nuclides due to the detection limit of the ICP was applied. From the Freundlich and Langmuir adsorption isotherms, the adsorption of the nuclides differs according to the concentration of the nuclides. As the leachability index of the nuclides is affected by the adsorption of the nuclides on the binding material, the effect of nuclide concentration is expected to be not ignorable. Therefore, the leachability index difference according to the nuclide concentration should be compared to avoid over- or underestimation of the leachability index.

In this study, the difference in the leachability index according to the concentration of nuclides is aimed to be checked. Cs, Sr, and Co, which should satisfy the leachability index criterion in the WAC of the Gyeongju repository, were selected as target nuclides. Three concentrations were selected to compare the leachability index: 0.1 mol, 0.001 mol and below the regulatory exemption concentration. The concentration of non-radioactive nuclides in the leachant was measured by ICP-OES and ICP-MS while the concentration of radionuclides was measured by HPGe. The result of this study can be applied as background data enhancing the WAC or disposal concentration limit of the radionuclides in Gyeongju LILW repository.

Acknowledgements

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Keywords: Adsorption, Leachability index, Leaching test method, Radionuclide, Waste acceptance criteria

~~A Review of Cellulose Degradation Environment and Conditions~~

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Withdrawal

Radiation Shielding Effects of Tungsten and Boron Carbide Coating Materials Fabricated by HVOF Techniuqe

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This study focuses on the development of coatings designed for storage containers used in the management of radioactive waste. The primary objective is to enhance the shielding performance of these containers against either gamma or neutron radiation. Shielding against these types of radiation is essential to ensure the safety of personnel and the environment.

In this study, tungsten and boron cabide coating specimens were manufactured using the HVOF (High-Velocity Oxy Fuel) technuqe. These coatings act as an additional layer of protection for the storage containers, effectively absorbing and attenuating gamma and neutron radiation. The fabricated tungsten and boron carbide coating specimens were evaluated using two different testing methods.

The first experiment evaluates the effectiveness of a radiation shielding coating on cold-rolled steel surfaces, achieved by applying a mixture of WC (Tungsten Carbide) powders. WC-based coating specimens, featuring different ratios, were prepared and preliminarily assessed for their radiation shielding capabilities. In the gamma-ray shielding test, Cs-137 was utilized as the radiation source. The coating thickness remained constant at 250 μm . Based on the test results, the attenuation ratio and shielding rate for each coated specimen were calculated. It was observed that the gamma-ray shielding rate exhibited relatively higher shielding performance as the WC content increased. This observation aligns with our findings from the gamma-ray shielding test and underscores the potential benefits of increasing the tungsten content in the coating.

In the second experiment, a neutron shielding material was created by applying a 100 μm -thick layer of B₄C (Boron Carbide) onto 316SS. The thermal neutron (AmBe) shielding test results demonstrated an approximate shielding rate of 27%. The thermal neutron shielding rate was confirmed to exceed 99.9% in the 1.5 cm thick SiC+B₄C bulk plate. This indicates a significant reduction in required volume.

This study establishes that these coatings enhance the gamma-ray and neutron shielding effectiveness of storage containers designed for managing radioactive waste. In the future, we plan to conduct a comparative evaluation of the radiation shielding properties to optimize the coating conditions and ensure optimal shielding effectiveness.

Acknowledgements

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Keywords: Radioactive waste storage, Tungsten carbide coating, Gamma radiation shielding, Boron carbide coating, Neutron radiation shielding

An Analyzing Technical Trends in Vacuum Thermal Desorption and Surface Oxidation for Spent Activated Carbon

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Activated carbon (AC), extensively used across various industrial sectors, serves as a sponge for different types of gases due to its porous carbon material. These gases are attracted to the carbon substrate via van der Waals forces. In nuclear power plants, AC is commonly used to adsorb radioactive gases such as ^{86}Kr and ^{134}Xe , as well as radioiodine sources like ^{131}I and ^{133}I from gaseous effluents. Even if the adsorbed radioactive gases and radioiodine decay into non-radioactive elements, the spent AC still contains radioactive species with long half-lives, such as ^3H (Tritium, T) and ^{14}C (radiocarbon). Minimizing and separating waste that contains long-lived nuclides (e.g., ^{14}C) are pivotal components of an efficient waste management approach. A challenging aspect of effectively managing disposed AC is to minimize long-lived radioactive substances by eliminating them. This paper explores and summarizes the technology used to remove pollutants (^3H , ^{14}C) trapped within the pores of Activated carbon through thermochemical vacuum and surface oxidation processes. By recycling and reusing spent Activated carbon, we anticipate a reduction in the volume of radioactive waste, leading to decreased disposal costs. Furthermore, this paper will contribute as a valuable reference in future studies, enhancing the understanding of vacuum thermal desorption and surface oxidation of used Activated carbon.

Keywords: Radioactive carbon, Tritium, Spent activated carbon, Thermal desorption, Surface oxidation

Review of Radiation Risk Assessment Codes for Domestic Radioactive Waste Maritime Transportation

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At the end of 2022 there were 439 nuclear power reactors in operating around the world, including 25 nuclear power reactors of Korea. Domestic nuclear power plants (NPPs) continuously produce low and intermediate-level radioactive waste (LILW) and spent nuclear fuel (SNF). As amount of radioactive waste is increasing and interim storage facilities meet limitation of their capacity, radioactive waste need to be transported. Consequently, the demand for radioactive waste transportation is increasing and the importance of Radiation Risk Assessment Codes (RRACs) for radioactive waste transportation is also on the rise.

Considering the domestic situation where all NPPs are located on seaside, the radioactive waste transportation by ship is inevitable and the its risk assessment is very important for safety. Although various researches on radioactive waste transportation risk assessment is being actively conducted, research on domestic radioactive waste maritime transportation is insufficient.

In this study, MARINRAD and KM-RAD were used to review on the radioactive waste transportation risk assessment. The result of reviewing shows that MARINRAD used SNF as transporting radioactive materials and ‘SAND87-7067 (1987)’ as nuclide database, whereas KM-RAD used LILW and ‘IAEA Technical Report Series-422 (2004)’. To complement these limitations, we present an modernized integrated database by updating data and covering the radioactive materials from LILW to SNF. These results are expected to contribute to the development of RRACs for domestic radioactive waste maritime transportation.

Acknowledgements

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Keywords: Radioactive waste, Marine transportation, Risk assessment, Evaluation model, Low and intermediate level waste

Preliminary Disposal Safety Assessment of Activated Carbon Waste Used as Filtering Material

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Activated carbon (AC) is used for filtering organic and radioactive particles, in liquid and ventilation systems, respectively. Spent ACs (SACs) are stored till decaying to clearance level before disposal, but some SACs are found to contain C-14, a radioactive isotope with 5,730 years half-life, at a concentration greater than clearance level concentration, 1 Bq/g. However, without waste acceptance criteria (WAC) regarding SACs, SACs are not delivered for disposal at current situation. Therefore, this paper aims to perform a preliminary disposal safety examination to provide fundamental data to establish WAC regarding SACs

SACs are inorganic ash composed mostly of carbon (~88%) with few other elements (S, H, O, etc.). Some of these SACs produced from NPPs are found to contain C-14 at concentration up to very-low level waste (VLLW) criteria, and few up to low-level waste (LLW) criteria. As SACs are in form of bead or pellets, dispersion may become a concern, thus requiring conditioning to be indispersible, and considering VLL soils can be disposed by packaging into soft-bags, VLL SACs can also be disposed in the same way, provided SACs are dried to meet free water requirement. But, further analysis is required to evaluate radioactive inventory before disposal.

Disposability of SACs is examined based on domestic WAC's requirement on physical and chemical characteristics. Firstly, particulate regulation would be satisfied, as commonly used ACs in filters are in size greater than 0.3 mm, which is greater than regulated particle size of 0.2 mm and below. Secondly, chelating content regulation would be satisfied, as SACs do not contain chelating chemicals. Also, cellulose, which is known to produce chelating agent (ISA), would be degraded and removed as ACs are produced by pyrolysis at 1,000°C, while thermal degradation of cellulose occurs around 350~600°C. Thirdly, ignitability regulation would be satisfied because as per 40 CFR 261.21, ignitable material is defined with ignition point below 60°C, but SACs has ignition point above 350°C. Lastly, gas generation regulation would be satisfied, as SACs being inorganic, they would be targeted for biological degradation, which is one of the main mechanism of gas generation.

Therefore, SACs would be suitable to be disposed at domestic repositories, provided they are securely packaged. Further analysis would be required before disposal to determine detailed radioactive inventories and chemical contents, which also would be used to produce fundamental data to establish WAC.

Acknowledgements

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Keywords: Activated carbon, WAC, Radioactive waste, Safety assessment

Simple Tests to Verify Decommissioning Waste Inventory at an NPP for Various Hypothetical Scenarios

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Kori Unit 1 was permanently shut down in 2017 and is currently being prepared for decommissioning. Decommissioning waste generated during the decommissioning of a nuclear power plant has the characteristic of being generated in large quantities over a short period. Therefore, if proper management is not carried out, abnormal situations (i.e., unauthorized disposal, diversion, etc.) may occur. According to IAEA General Safety Report Part 6, radioactive waste shall be managed for all waste streams in decommissioning. This means ensuring that all waste streams are managed by the recorded inventory of all decommissioning waste and verifying that the recorded inventory is reasonable. The radioactive waste management has been managed in units such as mass and radioactivity. However, in the case of decommissioning waste, the amount is very large, so management by radioactivity is expected to have limitations. Therefore, in this study, a simple test was conducted to verify the decommissioning waste generated by a hypothetical scenario by mass.

In this study, establish a scenario assuming various flows of decommissioning waste expected to be generated and calculate the expected inventory of decommissioning waste using Microsoft Excel. Specifically, using “Material Unaccounted For” (MUF), a material balance equation in IAEA Services Series 15, Nuclear Material Accounting Handbook, the error inventory was calculated as the difference between the physical inventory of decommissioning waste in the area and the ending inventory. We propose a simple test scenario to verify the flow of decommissioning waste by verifying that the error inventory reasonably matches the set allowable error.

This study aims to verify the inventory of decommissioning waste using the material balance methodology used for nuclear material accounting. It is expected that the safety and reliability of the nuclear power plant decommissioning process can be secured by verifying that the total inventory of equipment before decommissioning and the inventory of remaining equipment and decommissioning waste after decommissioning are reasonably consistent.

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Keywords: Decommissioning, Waste management, Inventory verification, Material balance

Validation of ISOCS Measurements Using MicroShield

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In nuclear power plant environments, the analysis of gamma-emitting waste materials with complex shapes can be challenging. ISOCS (In-Situ Objective Counting System) is employed to measure the gamma-emitting radionuclide concentrations. However, it is crucial to validate the accuracy of ISOCS measurements. This study aims to validate the accuracy of ISOCS measurement results for spent filters.

The ISOCS measurement process begins with modeling and efficiency calculations of the target spent filters using ISOCS software. ISOCS offers the advantage of direct measurement assessment by incorporating shielding materials and collimators into the detector efficiency calculation during the modeling process, without the need for separate efficiency correction sources. To validate the accuracy of ISOCS measurement results, the measured radioactivity values were used as input data for the MicroShield computer code to derive dose rates. These dose rates were then compared to the dose rates measured on-site, confirming the reliability of ISOCS measurements.

In the field, ISOCS gamma measurements and surface dose rates were measured for three Cavity filters and four RCP Seal Injection filters. The measured dose rate for the Cavity filters was around 270 $\mu\text{Sv/hr}$, and the computed values using MicroShield showed an error of approximately 12%. Despite modeling and calculation errors in computer analysis and potential uncertainties in the measurement environment and instrument, the computed values closely matched the measured values.

However, the measured dose rate for the RCP Seal Injection filters ranged 2.9~8 $\mu\text{Sv/hr}$, which is very low and close to background levels. When compared to the results of computer analysis, an error ranging from 27% to 97% was observed. It is concluded that validating the accuracy in the low dose rate range close to background levels is challenging through a comparison of calculated and measured dose rates.

Keywords: Spent filter, ISOCS, MicroShield

Development of a Remote Processing System for Spent Filter Handling

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Since 1996, spent filters from the Kori unit 1 have been stored in enclosed areas such as the auxiliary building filter room. To dispose of these spent filters at a disposal facility, it is necessary to retrieve and package them according to the disposal criteria. The Kori unit 1 filter room is a 2.5-meter deep hole with 227 spent filters stored indiscriminately by type and radiation level. Furthermore, the exposure dose rate measurements revealed exceed 10 mSv/h, making it a challenging environment for workers. Therefore, in this study, we have developed a 'Remote Processing System for Spent Filter Handling' to minimize worker exposure and ensure safety throughout the entire process, from filter retrieval to radiation measurement, sample collection, compression, and packaging. We have completed performance testing through laboratory validation.

The 'Remote Processing System for Spent Filter Handling' consists of four main components: a robot system for retrieving spent filters from the filter room, a transfer mechanism for moving spent filters to the lower area, a core ring device for sample collection, and finally, a compression/packaging unit. The laboratory validation performance testing was conducted by installing these devices in a structure simulating the Gori-1 reactor filter room. The results confirmed that all processes, from spent filter retrieval to packaging, can be remotely operated without the need for filter drops or worker intervention.

Through the laboratory validation, some areas for improvement were identified. These improvements should be taken into consideration when producing the system for future on-site applications.

Keywords: Spent filter, LILW, Disposal, Remote processing system

Economic Considerations on Low Level Radioactive Waste (LLW) Management and Appropriate Approaches as an Engineer

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Every engineering decision in radioactive waste management should be based on both technical and economic considerations. Especially, the management of low-level radioactive waste (LLW) is more critical on economic concerns, due to its long-term and continuous nature, which emphasizes the importance of economic analysis. In this study, economic factors for LLW management were discussed with appropriate engineering applications. Two major factors that should be taken into account when assessing economic expectations are the accuracy of the results and its proper balancing with ALARA philosophy (As Low As Reasonably Achievable).

The accuracy of the results depends on the correct application of alternatives within a realistic framework of waste processing. This is because the LLW management process involves variables such as component type, physical dimensions, and the monetary value at the processing date. Two commonly used alternatives are the simplified lump sum present worth and levelized annual cost methods, which are based on annual and capital costs. However, these discussions on alternatives not only pertain to the time series value of operational costs but also to future technical advancements, which are crucial for engineers. As new research results on LLW treatment emerge, proper consideration and adoption should be given to technical cost management.

As safety is the core value of the entire nuclear industry, the ALARA philosophy should also be considered in the cost management of LLW. The typical cost of exposure in man-rem has ranged from \$1,000 to \$20,000 over the past decades. However, with increasing concerns about health and international political threats, the cost of man-rem should be subject to stricter criteria, even the balancing of costs and safety concerns is much controverse issue.

Throughout the study, the importance of incorporating proper engineering insights into the assessment of technical value for the financial management of LLW was discussed. However, it's essential to remember that financial management should not be solely assessed based on the size of expenses but rather by evaluating the current financial status, the value of money at the time, and anticipated future costs, considering the specific context and timeframe.

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Keywords: Low level radioactive waste, Economic consideration, Engineering finance

Fabrication of a Cement Waste Form for Final Disposal of Spent Ion Exchange Resins

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Spent ion exchange resins have been generated during the operation of nuclear facilities. These resins include radioactive nuclides. It is needed to fabricate them into a stable form for final disposal. Cement solidification process is a useful method for the fabrication of them into a waste form for final disposal. In this study, proper conditions for the fabrication of them into a stable waste form were determined using the cement solidification process. In-drum waste forms were then produced at the conditions, where the stability of representative samples was evaluated for final disposal. The samples were satisfied to the Waste Acceptance Criteria for low and intermediate level radioactive waste disposal sites. This result can be utilized to derive optimal conditions for the fabrication of spent ion exchange resins into a final disposal form.

Keywords: Spent ion exchange resin, Final disposal, Cement solidification, In-drum waste form

A Study of Overseas Immobilization Requirements for Heterogeneous Radwaste

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In general, radioactive waste with high radioactivity is made into a solid form with performance such as leaching restriction, shape retention, and structural stability so that radioactive waste does not affect humans and the environment as much as possible. This should be applied equally to radioactive waste, whether homogeneous or heterogeneous. The requirements are stipulated in the “Low and Intermediate Level Radioactive Waste Delivery Regulations” notice of the Korea Nuclear Safety and Security Commission. On the other hand, the waste acceptance criteria for domestic disposal facilities require immobilization of heterogeneous waste when the activity concentration is above a certain level, but do not provide specific immobilization performance requirements.

In this study, the immobilization requirements applied to heterogeneous radioactive waste in various overseas countries operating low and intermediate-level radioactive waste disposal facilities were studied. First, the IAEA’s safety standards for radioactive waste immobilization, domestic regulations, and disposal facility waste acceptance criteria were reviewed. Countries operating surface disposal facilities such as the United States, France, Spain, and Japan and countries operating underground disposal facilities such as Sweden and Finland were divided to review the current status of immobilization application to heterogeneous waste in overseas countries. When reviewing overseas cases, each country’s disposal methods, types of disposal waste, and waste treatment criteria were also reviewed. It was found that the immobilization requirements for heterogeneous radioactive waste vary depending on the disposal method and the type of barrier used to ensure disposal safety in each country. The common point is to surround heterogeneous radioactive waste within a concrete lining of a certain thickness, and to apply the thickness, compressive strength, and diffusion coefficient of the concrete lining as immobilization performance requirements.

Through this study, the immobilization performance requirements for heterogeneous radioactive waste in various overseas countries that stably operate low- and intermediate-level radioactive waste were confirmed, which is expected to contribute to specifying the performance requirements for immobilization of heterogeneous radioactive waste in domestic disposal facilities.

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Keywords: Low and intermediate level, Waste acceptance criteria, Heterogeneous radioactive waste, Immobilization, Requirement

A Study on Plasma Torch Melting According to Waste Type and Condition

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Republic of Korea is preparing to decommission Kori Unit 1 and Wolsong Unit 1. Decommissioning of a nuclear power plant proceeds in the following stages: shutdown, transition period, decontamination, cutting, waste treatment, and site restoration. When nuclear power plant is decommissioned, It is expected that approximately 80,000 drums of radioactive waste will be generated per nuclear power plant. Therefore, various technologies are being researched and developed to reduce this to approximately 14,500 drums.

Technologies for waste volume reduction are largely mechanical and electrical/thermal methods. Representative examples of mechanical volume reduction technologies include super compactors and electrical/thermal volume reduction technologies include induction and plasma torch furnaces. Both technologies are effective reduction technologies, but the reduction ratio varies depending on the type or condition of waste before treatment. For example, as a result of testing waste reduction using a super compactor at NUKEM in Germany, the reduction ratio was found to be between 1.3 and 7 depending on the type or condition of waste such as chips, ash, scrap metal, sand, etc. And according to IAEA-TECDOC-1527, when reducing the volume of metals, aluminum, lead, copper, brass, etc. using induction melting, the waste volume reduction ratio is 5 to 20.

In this paper, referring to these results, a melting test was conducted using a previously developed plasma torch with an output of more than 100 kW. And volume reduction characteristics of this plasma torch was considered depending on waste type or condition.

Keywords: Plasma, Melting, Radioactive waste, Volume reduction, Waste treatment

Leaching Characteristics of Cs(I) and Co(II) Immobilized in Cement Waste Forms Containing EDTA

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Radionuclides in low- and intermediate-level radioactive wastes from the decommissioning process of nuclear power plants were generally immobilized by cementation methods. Ethylenediaminetetraacetic acid (EDTA), which is extensively used as a decontamination agent, can affect the behaviors of radionuclides immobilized in cement waste forms. In this study, the effects of EDTA contained in simulated radioactive decommissioning wastes on the leaching characteristics of immobilized Co and Cs and the microstructure evolution of cement waste form. Co leaching was accelerated by the formation of Co–EDTA complexes with high mobility and solubility. Cs leaching was hindered by the ion competition with other metal–EDTA complexes for releasing from the cement waste form. Cs leaching was also retarded by carbonated layer at edge of the cement waste form, which process is facilitated by the presence of EDTA. Finally, the effects of EDTA on the leaching characteristics of immobilized Cs and Co and the microstructure evolution of the cement waste form should be considered to ensure the safety of disposal for low- and intermediate-level radioactive wastes.

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Keywords: Cesium, Cobalt, Leaching, EDTA, Cementation, Carbonation

Space Planning of Radionuclide Analysis Facility for Below Low-Level Decommissioning Radioactive Wastes at KRID

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Recently, the nuclear decommissioning and environmental restoration industries has significantly attracted as a new industry field due to the decision to decommission the KORI#1 and WOLSONG #1 nuclear power plant. In order to dispose of the decommissioning radioactive wastes generated during nuclear decommissioning, proper analysis is required, and disposal decisions are determined based on the analysis results. When dismantling a nuclear power plant, a few thousand of tons decommissioning waste are produced, so these require analysis for proper disposal. Therefore, a radionuclide facility for decommissioning waste analysis is essential for the disposal of the large quantities of decommissioning waste generated during nuclear power plant decommissioning.

Korea Research Institute of Decommissioning (KRID) was established radionuclide analysis facilities to address above issues and support nuclear power plant decommissioning projects. The plan is to perform classification by type and radionuclide for all waste produced during nuclear power plant decommissioning and to support the disposal of radioactive wastes. In addition, we plan to establish validation methods for samples where verification methods are not established, in order to conduct efficient analysis and management. In this presentation, we will introduce the radionuclide facility currently under construction at KRID and present the space design, equipment layout, and utilization plans.

Keywords: Decommissioning, Radionuclide analysis, R&D project, KRID

Proposal of Treatment Process for Concrete Repackaging Container of Concentrated Effluent and Spent Resin

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Concentrated effluent and spent ion exchange resins (IERS) from nuclear power plants (NPPs) were generated prior to the establishment of a disposal facility site and waste acceptance criteria have been temporarily stored at the NPPs because their suitability for disposal has not been confirmed. In particular, at the Kori Unit 1, which was the first to start the commercial operation in South Korea, the initially generated concentrated effluent and IERS are repackaged in large size of concrete containers and stored without provided regulation standard. The concentrated effluent is package as cementitious form in 200 L drums and repackaged in concrete containers, case of the IERS were solidified or dehydrated and repackaged in round concrete container.

In this study, we review and propose a disposal plan for concentrated effluent and IERS repackaging drums that have not been confirmed to be suitable for disposal from the first operating nuclear power plant, Kori Unit 1, 2.

First, the concentrated effluent was stored in four 200 L drums respectively, and then, it was again stored in concrete container and which was poured on top using grouted concrete. Therefore, the process was required by cutting concrete container for extracting the internal drums at first. Internal radioactive waste should be crushed to the suitable waste criteria and solidified, finally disposal in to the polymer concrete high integrity container (PC-HIC).

IER was repackaged and disposal in square type of 200 L concrete drums respectively covered the cap. So, extracting the internal drums should be extracted after removing the cap of external concrete container. Cement solidification drums can be crushed and re-solidified or disposed in the PC-HIC. Stored IER after dehydrated can be disposal in PC-HIC.

In conclusion, the container was used as a package that repackaging the concentrated effluent and IER was separated into two different types of waste depending on the level of contamination of radioactivity, the polluted area is disposed of as radioactivity contamination or the unspoiled area will be treated as self-disposal waste.

Keywords: Concentrated effluent, Spent ion Exchange resin, Concrete repackaging drum

Characteristic Analysis of DAWs to Derive Heat Treatment Process Conditions

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Dry active wastes (DAWs) are combustible waste generated during the operation and decommissioning of nuclear facilities, and are known to be generated in the amount of approximately 10,000 to 40,000 drums (based on 200 L) per unit. It consists of various types of protective clothing, paper, and plastic bags, and is stored in radioactive waste storage facilities. Therefore, reducing the volume of DAWs is an important issue in order to reduce storage costs and utilize the limited space of waste storage facilities.

Heat treatment such as incineration can dramatically reduce the volume of waste, but as the waste is thermally decomposed, CO₂, a global warming gas, is generated and there is a risk of emissions of harmful gases including radionuclides. Therefore, a heat treatment process that minimizes the generation of CO₂ and harmful gases is necessary. One of the alternatives to incineration is to carbonize DAWs, dispose of carbonized materials below the release standard as non-radioactive waste, and selectively separate and stabilize inorganic components, including radionuclides, from carbonized DAWs.

In this study, 13 types of DAWs generated from nuclear power plants were selected and their thermal decomposition characteristics were investigated to design a heat treatment process that replaces incineration. As a result of TGA analysis, the temperature at which thermal decomposition of each waste begins is 260-300°C for cotton, 320-330°C for paper, 315-420°C for synthetic fiber, 350°C for latex gloves. The mass of most samples decreased to less than 1 % of the initial weight after heat treatment, and dust suit and latex gloves had residues of 13.83% and 13.71% of the initial mass, respectively. The metal components of the residue produced after heat treatment of the sample were analyzed by EDS. According to the EDS results, cotton contains Ca and Al, paper contains Ca, Al and Si, synthetic fiber contains Ca, Cu and Ti, latex gloves contain Ca and Mg. Additionally, ICP analysis was performed to quantify the inorganic components. These results are expected to be applicable to the processing of DAW generated at nuclear facilities in the future.

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Keywords: Dry active waste, Alternative incineration, Thermal decomposition, Volume reduction

Effect of Leachant Type on the Leaching Behavior of Technetium and Iodine Inside the Cement Solidified Waste

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Wasteform is the first barrier to prevent radionuclide release from repositories into the biosphere. Since leaching rates of nuclides in wasteform significantly impact on safety assessment of the repository, clarifying the leaching behavior is critical for accurate safety assessment. However, the current waste acceptance criteria (WAC) of the Gyeongju repository only evaluates leachability indexes for Cs, Sr, and Co, which are tracers for nuclear power plant waste streams. Furthermore, ANS 16.1, the current leaching test method used in WAC, applies deionized water (DI) as leachant. However, the interactions between wasteform and groundwater environment in the repository may not be reflected. Therefore, it is necessary to review the current leaching test method and nuclides that may require the extra evaluation of leachability beyond the Cs, Sr, and Co.

Tc and I are key nuclides contributing to high radioactive dose in safety assessment due to their high mobility and low retardation factor. The groundwater conditions within the repository, such as pH and Eh significantly affect the chemical form of Tc and I. For example, Tc in H₂O system tends to form hydroxide precipitates in neutral pH condition and TcO₄⁻ in strong alkaline environments according to the Pourbaix diagram. In case of I, it generally exists in the form of I⁻, while it exists as IO₃⁻ as Eh increases. Although the current leaching test at the Gyeongju repository applies DI as a leachant, the actual repository is expected to have a highly alkaline environment with a substantial amount of various ions in the groundwater. Consequently, the leaching behavior in the ANS 16.1 test and the actual disposal condition is different. Thus, it is necessary to analyze the leaching behavior of Tc and I with reflecting the actual disposal environment.

In this study, the leaching behavior of Tc and I is investigated by following ANS 16.1 leaching test method. The solidified waste specimens containing 10 mmol of Re and I were manufactured with cement, which is widely used as a solidification material. Re was applied instead of Tc, which has similar chemical behavior to Tc, and NH₄ReO₄ and NaI were used as surrogates for Re and I. As a leachant, deionized water and cement-saturated groundwater were prepared and the concentration of nuclides in the leachant is analyzed by ICP-OES. As the result of this study, experimental data can be applied to improve the WAC and disposal concentration standards in the future.

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Keywords: Leachability index, Leachant, Portland cement, Radioactive waste, Solidified waste

Development Technology of Concrete Container Characteristic Evaluation

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At domestic nuclear power plant, concrete containers are stored to store waste generated before waste acceptance criteria (WAC) was established. Concrete container store concentrated waste liquid and waste resin. In order to disposal radioactive waste to a disposal site, it is necessary to conduct a characteristic evaluation inside the waste to check whether it satisfies the WAC.

Two types of concrete containers are stored: round and square. The round type is filled with one 200-liter drum, and the square type is filled with four 200-liter drums. In the case of a round shape, the top lid is fastened with bolts, so it is possible to collect samples after opening the top lid without the need for additional equipment. However, in the case of a square shape, there is no top lid, and concrete is poured to cure the lid, so the separate equipment for characteristic evaluation is required. It is necessary to install a workstation for sample collection on the top of the concrete container, equipment for coring the top of the concrete container, and a device to prevent concrete dust scattering.

Currently, the design of equipment for evaluating the characteristics of concrete containers has been completed, and equipment optimization through mock-up test will be performed in the future.

Keywords: Concrete container, Characteristic evaluation, Waste acceptance criteria

A Case Study on Clearance Dose Assessment of Radioactive Waste at KRR 1&2

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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 expected that more than 5,000 tons of dismantled wastes will be generated as the contaminated building is demolished. Since these dismantled wastes must be disposed of in an efficient method considering economic feasibility, it is desirable to clearance extremely low-level wastes whose contamination is so minimal that the radiological risk is negligible. In Korea, in order to approve the clearance of radioactive waste, it must be proven that the nuclide concentration standards are met or that the dose to individuals and collectives is below the allowable dose value. At the KRR 1&2 decommissioning site, dismantled wastes have been steadily being disposed of through clearance procedure since 2021. Clearance was approved by the Korean Institute of Nuclear Safety (KINS) for one case of concrete waste in 2021 and two cases of metal waste in 2022. In 2023, the clearance of metal waste and asbestos waste has been approved so far, and in particular, this is the first case in Korea for asbestos waste.

In this study, we compared the dose assessment methods and results of clearance wastes at the KRR 1&2 decommissioning site from 2021 to present. Dose assessment was conducted by applying the landfill scenario for concrete and asbestos and the recycling scenario for metal waste. The calculation codes used were RESRAD-onsite 7.2 and RESRAD-recycle 3.10. The dose conversion factors (DCF) for each age group (infant, 1y, 5y, 10y, 15y, adult) of the target nuclide used the values presented in ICRP-72, and in particular, geo-hydrological data of the actual landfill site was used as an input factor when evaluating landfill scenarios. As a result of the dose assessment, when landfilling concrete wastes in 2020, the personal dose and collective dose were evaluated the most at $2.80\text{E}+00 \mu\text{Sv/y}$ and $4.83\text{E}-02 \text{man}\cdot\text{Sv/y}$, respectively.

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Keywords: Clearance, Dose assessment, Decommissioning, KRR 1&2, Radioactive waste

Criticality Safety Analysis for Concrete Radioactive Waste Drums Using KENO-VI of SCALE

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Most of the radioactive wastes generated during the nuclear fuel processing activities conducted by KEPCO Nuclear Fuel Co., Ltd. are classified as the categories of intermediate and low-level radioactive waste. These radioactive waste materials are intended for permanent disposal at a designated disposal site, adhering strictly to the waste acceptance criteria. To facilitate the safe transportation of radioactive waste to the disposal site, it is necessary to ensure that the waste drums maintain a level of criticality that complies with the waste acceptance criteria. This necessitates the maintenance of subcritical conditions, under immersion or optimal neutron moderation conditions.

This paper presents a criticality safety assessment of concrete radioactive waste under the most conservative conditions of immersion and moderation conditions for waste drums. Specifically, In order to send radioactive waste, which is the subject of criticality analysis, to a disposal facility, pre-processing operations must be performed to ensure compliance with waste acceptance criteria. To meet the physical characteristics required by the acceptance criteria, particles below 0.2 mm should not be included. Thus, a 0.3 mm sieve is used to separate particles larger than 0.3 mm, and only those particles are placed in drums. The drums should be filled to achieve a filling ratio of at least 85%.

A criticality analysis was conducted using the KENO-VI of SCALE. The Criticality Safety Analysis Results of varying the filling ratio of concrete drums from 85% to 100% presented in an effective multiplication factor of 0.22484. Additionally, the effective multiplication factor presented to be 0.25384 under the optimal moderation conditions. This demonstrates full compliance with the USL and criticality technology standards set as 0.95.

Acknowledgements

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Keywords: Criticality safety, Radioactive waste, Upper safety limit, SCALE

Comparison of Oversea Plasma Torch Melting Facility for Radioactive Waste

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Plasma torch melting technology has been considered as a promising technology for treating or reducing the radioactive waste generated by nuclear power plants. In 2006, IAEA announced that the technology is able to treated regardless of the type of target wastes. Because of the advantage, many countries have been funding, researching and developing the treatment technology. In this study, overseas plasma torch melting facilities for radioactive wastes treatment are reviewed. Also, plasma torch melting facility developed by KHNP CRI is briefly introduced.

Keywords: Plasma torch melting facility, Treatment technology, Oversea, Review

Review on the Methodology for Evaluating the Radioactivity in the High Integrity Container Packages of Dried Spent Resin and Concentrate Radwaste

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To safely dispose of highly radioactive spent resin and concentrate waste generated through nuclear power plant operations, it is essential to meet the physicochemical properties requirements of the packages and ensure the accuracy and reliability of radiological characteristics determination. Both spent resin and concentrate are packaged in high-integrity containers (HICs) after drying and are homogeneous waste products generated in the primary system and liquid radioactive waste treatment system. Meeting the physicochemical properties requirements does not appear to be difficult. However, to achieve reliable radiological characterization of high-integrity container packages, it is necessary to take a representative sample and perform accurate radiological analysis. Therefore, this paper discusses the methodology for evaluating the radionuclide inventory of high radioactive resin and concentrate packages, as well as the essential element technology and considerations. For relatively high radioactive resin and concentrate packages, the radionuclide inventory for each package should be evaluated with high reliability through direct radiological analysis of the representative samples collected for each package. This can contribute to the efficient operation of radioactive waste disposal facilities. Radionuclide-specific concentrations directly analyzed for each package will be managed in a database. As analytical data accumulates and direct measurements of high-integrity container package such as the radwaste drum assay system (RAS) become feasible, statistical techniques such as correlation analysis between easy-to-measure (ETM) nuclides and difficult-to-measure (DTM) nuclides can lead to the development of efficient and reasonable indirect evaluation methods, such as scaling factor and the mean activity concentration method. As for the element technology, a remote representative sampling technique should be developed to safely and effectively take representative samples of highly radioactive materials, including granulated or hardened concentrate waste. Considerations should also be given to determining the sample quantity representing each package, as well as establishing radiation calibration and measurement methods appropriate to the radiation levels of the representative samples.

Keywords: Remote sampling technique, HIC package, Representative sample quantity

A Case Study on Assessment Scheme for Radioactivity Inventory of Dismantled Metal Waste From Japan Power Demonstration Reactor (JPDR) for Disposal

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For the disposal of radioactive waste from nuclear facilities, assessing their radioactivity inventories is essential. As a result, countries with nuclear facilities are implementing assessment schemes tailored to their respective policies and available resources for radioactive waste management. This paper specifically describes the assessment scheme for radioactivity inventory applied to metal waste generated during the dismantling of the Japan Power Demonstration Reactor (JPDR), a 1.25 MW BWR. The distinctive aspect of the Japanese approach lies in the fact that, for a pair of a key nuclide and a difficult-to-measure (DTM) nuclide that lack a significant correlation in their concentrations, the mean activity concentration method was used. In this method, an arithmetic average of all measurements of the DTM nuclide from representative drums, including MDAs (Minimum Detectable Activities), was assigned to the concentration of the DTM nuclide for all drums, regardless of the concentration of its paired key nuclide. Conversely, for a specific pair of a key nuclide and a DTM nuclide with a significant correlation, the scaling factor method was applied, as is common in many other countries. This Japanese case can serve as a valuable reference for Korea, which does not have the option of using the mean activity concentration method in its assessment scheme.

Keywords: Radioactive waste, Disposal, Radioactivity inventory, Mean activity concentration method, Scaling factor method, JPDR

Measuring System Configuration Plan for Determining Clearance Level

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Large amounts of concrete, metal, soil, and other radioactive waste are generated not only from nuclear power plants operating in Korea but also from nuclear power plant decommissioning. If it is confirmed through measurement of residual radioactivity that the concentration is below the allowable clearance level, they can be managed as general or industrial waste in accordance with the Nuclear Safety Act.

The Korea Radioactive Waste Agency predicts that very low-level radioactive waste will be generated the most, at about 67.1%. If waste below clearance level among very low-level radioactive waste can be evaluated and reduced, a lot of costs can be saved.

Among radioactive wastes, metal wastes in particular have various sizes, shapes, and densities.

If radioactivity is measured without properly considering this, a large error occurs in the measured value even if the radioactivity value is the same. This requires a conservative measurement method using density correction taking into account the self-absorption effect.

For conservative measurements, it is essential to compare measured values with calculated values using MCNP6 (Monte Carlo N-Particle). You must enter the geometry of the measurement environment and derive calculated values using F8 Tally. Clearance level of radioactive waste is determined through the above method.

In addition, sufficient MDA (Minimum Detectable Activity) must be secured to determine clearance level by using NaI(Tl), plastic scintillator configuration, and lead shielding. Nuclide analysis is performed using a NaI(Tl) scintillator and the total gamma radioactivity is evaluated using a highly efficient plastic scintillator.

Keywords: Decommissioning, Clearance level, Radioactive waste, Measurement

Characteristics of Paraffin Solidification Drum With Boron-Containing Radioactive Liquid Waste Dry Powder

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Radioactive liquid waste generated during the operation of domestic nuclear power plants is treated through a somewhat different liquid radwaste system (LRS) for each plant. Prior to the introduction of standard nuclear power plants, LRS used a concentrated water dry system (CWDS) to evaporate liquid waste and manage it in the form of dry powder. The boron-containing radioactive liquid waste dry powder was solidified using paraffin from 1995 to 2010, and about 3,650 drums (based on 200 L) of paraffin solidified drums are currently stored in nuclear power plants. Paraffin solidification drums do not meet the acceptance criteria for radioactive waste repositories because it is difficult to secure the homogeneity of the solidified body and there are concerns about leaching of radioactive waste due to the low melting point of paraffin. In order to solve this problem and safely permanently dispose of paraffin solidification drums, the characteristics of dry powder paraffin solidification drums containing boron-containing radioactive liquid waste must be analyzed and appropriate treatment technology utilizing the results must be introduced.

This study analyzes the physical properties of paraffin, the chemical properties of boron-containing radioactive waste dry powder, and the physicochemical properties of paraffin solidification powder, and proposes an appropriate alternative technology for treating boron-containing radioactive waste dry drum. When disposing of the paraffin solidification drum with boron-containing radioactive liquid waste dry powder, the solidification body must be effectively withdrawn from the drum and the paraffin must be completely separated from the solidification body. When disposing the drum, the solidified material must be effectively extracted from the drum and the paraffin must be completely separated from the solidified material. Afterwards, the paraffin must be self-disposed, and the radioactive waste must be disposed of in accordance with acceptance criteria of repository. We looked at how each characteristic of the paraffin solidification drum with boron-containing radioactive liquid waste dry powder can be utilized in each of the above treatment processes.

Acknowledgements

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Keywords: Liquid radwaste, Paraffin, Solidification, Boron-containing

Evaluation of the Solidification Characteristics of Dispersible Radioactive Waste Using Organic-Inorganic Hybrid Solidifying Binders

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Various types of solidifying materials are used to stabilize and solidify low and intermediate-level radioactive dispersible waste. Portland cement is generally used to solidify various radioactive wastes because its facilities and processes are simple, less dangerous, and it has excellent compressive strength after curing compared to other materials. However, it is difficult to use Portland cement in radioactive waste containing highly water-soluble harmful substances such as sodium fluoride because it is prone to leaching harmful ingredients in immersion tests due to its low water resistance. In this study, solidification was achieved using an organic-inorganic hybrid solidifying binders consisting of inorganic binders such as Portland cement, blast furnace slag powder, silica fume, and organic binders such as epoxy resin. This material was then compared with a solidification material made of Portland cement alone. The mixing ratio of inorganic binders, water, and organic binders to simulated waste is 35%, 20%, and 25%, respectively. The mixing ratio of inorganic binders and water when using only Portland cement for simulated waste is 100% and 80%, respectively. The mixed paste was poured into a cylinder mold ($\Phi 5 \times 10$ cm) to seal the upper part, cured at room temperature for 28 days to produce a solidification specimen, and then subjected to various tests were performed, including compressive strength, immersion compressive strength, hydration peak temperature, length change, and immersion weight change. The compressive strength of the organic-inorganic hybrid solidification test was 13-17 MPa, the immersion compressive strength was 15-18 MPa, the hydration peak temperature was 33-36°C, the length change rate was -0.086%, and the immersion weight change rate was -2.359%. The compressive strength of the Inorganic solidification test using only Portland cement was 16-18 MPa, the immersion compressive strength was 20-21 MPa, the hydration peak temperature was 23-25°C, the length change rate was -0.150%, and the immersion weight change rate was -5.213%. The compressive strength and immersion compressive strength of the organic-inorganic hybrid solidification materials were slightly lower compared to those of Portland cement solidification materials, they still met the compressive strength standard of 7-12 MPa, taking into consideration the strength reduce and economic feasibility of the core drill process. Furthermore, it indicates that the rates of change in length and immersion weight decreased to about 1% and 5%, suggesting an improvement in water resistance. The above results suggest that applying the organic-inorganic hybrid solidification method to radioactive waste treatment can effectively improve water resistance and help secure long-term stability.

Keywords: Dispersible radioactive waste solidification, Sodium fluoride, Inorganic-organic hybrid solidifying binder, Epoxy-cement binders, Compressive strength, Water resistance

Evaluation of Similarity in Radioactive Distribution Between Two Groups for Assessing the Inventory of Radionuclides in Nuclear Power Plant Waste

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To effectively assess the inventory of radionuclides generated from nuclear power plants using a consistent evaluation method across diverse groups, it is imperative to analyze the similarity in radioactive distribution between these groups. Various methodologies exist for evaluating this similarity, and the application of statistical approaches allows us to establish similarity at a specific confidence level while accounting for the dataset size (degrees of freedom). Initially, if the variance characteristics of the two groups are similar, a t-test for equal variances can be employed. However, if the variance characteristics differ, methods for unequal variances should be applied. This study delineates the approach for assessing the similarity in radioactive distribution based on the analytical characteristics of the two groups. Furthermore, it delves into the results obtained through two case studies to offer insights into the assessment process.

Keywords: Similarity, Freedom, Confidence level, Scaling factor, Radionuclide inventory

Validation of Correlation Coefficients for Indirect Estimation of Radionuclide Inventory

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In order to apply indirect methods (such as scaling factors) to assess the radionuclide inventory of waste generated by nuclear power plants, it is essential to first evaluate the correlation coefficient between key radionuclides and those that are difficult to measure (DTM). The benchmark for the correlation coefficient (r) applied in indirect assessments is set at 0.6, and its significance can vary based on both its value and the size of the dataset. For instance, deriving a correlation coefficient using three data points versus utilizing a dataset with a hundred data points would yield different implications. This study addresses the variance in correlation coefficients based on data selection and presents a methodology for validating the significance of these coefficients. Additionally, we will discuss how these variances may impact the results of indirect assessments, such as scaling factor evaluations.

Keywords: Correlation, Coefficient, Scaling factor, Radionuclide inventory

Review of Scaling Factors and Average Radioactivity Concentration Application Methods

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To evaluate the inventory of radionuclides for the disposal of waste generated from nuclear power plants, indirect assessment methods such as the scaling factor method or average radioactivity concentration method can be applied. A scaling factor represents the average concentration ratio between key radionuclides and difficult-to-measure (DTM) radionuclides, while the average radioactivity concentration refers to the average concentration of DTM radionuclides, regardless of the concentration of key radionuclides or within specific ranges of key radionuclide concentrations. These indirect assessment methods can be statistically derived through the analysis of representative drums. This study will address how to apply these scaling factors and average radioactivity concentrations. Firstly, the concentration of gamma-emitting radionuclides will be analyzed using a drum radionuclide analyzer, and the concentration of DTM radionuclides will be determined by applying scaling factors specific to each DTM radionuclide. In the case of using the average radioactivity concentration method, the average concentration of DTM radionuclides will be applied independently of the concentration of gamma-emitting radionuclides. It is crucial to perform radioactive decay correction based on the date of generation or disposal when applying scaling factors or average radioactivity concentration. Additionally, for repackaged 320 L drums, determining which drum among the two 200 L drums inside should serve as the reference is of utmost importance.

Keywords: Average radioactivity concentration, Scaling factor, Indirect assessment, Application, Radionuclide inventory

Development and Application Cases of Clearance Procedures for Special Waste Type

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The radioactive waste generated within radiation-controlled areas is classified and processed according to relevant laws and regulations based on contamination levels. In cases where such radioactive waste complies with the legally defined clearance concentration or dose criteria, it is disposed of as non-radioactive waste by means of incineration, reclamation, recycling, etc.

Within radiation controlled areas, various consumables are periodically replaced to ensure the proper operation of the area. It is necessary to have appropriate disposal methods for these consumables. In particular, waste items such as fire extinguishers, fluorescent lamps, batteries, and pressure vessels (hereinafter referred to as “Special Waste Type”), which may contain hazardous substances within their internal components and contents, should be considered for appropriate disposal methods that comply with nuclear safety and environmental laws.

In the present case, the specified special waste type do not come into direct contact with radiation sources, and they have impermeable surfaces, which significantly reduces the risk of external contamination infiltrating the interior. However, the current method of clearance is not suitable for these items (Typically, nuclear energy-related business operators are required to classify clearance target waste based on internal and external components and demonstrate compliance with the criteria. Nevertheless, for special waste type, it is difficult to separate and measure internal and external components within the radiation-controlled area).

In this case, the Clearance Procedure for special waste type applied to Korea Atomic Energy Research Institute was introduced. Additionally, we have extracted considerations for future domestic clearance of the type.

Keywords: Clearance, Special waste type, Radiation controlled area

Design and Operations of a Melting Furnace System for Radioactive Metal Wastes

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Various radioactive metal wastes are generated during operation and decommissioning of nuclear facilities. Radioactive metal wastes with complex geometries or volumetric contamination can be difficult to decontaminate and disposal costs may increase. To solve these problems, the radioactive metal wastes can be treated by melting method. In this study, we designed a melting furnace system of air induction melting type, which is widely utilized due to its advantages of good thermal efficiency, uniform heating and guaranteed safety for radioactive material. By utilizing the melting furnace system, volatile radionuclides existed in the base material can be captured in the form of gas or dust by the filter. The radionuclides whose chemical properties can easily form metal oxides present as slag. For this reason, the specific radioactivity of the base material can be reduced. Radionuclides that are difficult to transport to slag and dust are uniformly distributed in the base material. A dedicated power supply and a transformer were necessary to be included in the melting furnace system since the induction furnace uses high-frequency currents. In addition, a hood is placed on top of the furnace to capture fumes generated during melting, and additional hoods were installed around the furnace to remove airborne dust. In particular, a dust collection unit consisting of a cyclone and a HEPA filter were constructed to effectively collect dust containing radionuclides. During the melting process, the slag is removed and accumulated separately, and the ingot production system was designed to produce the ingot using molten metal. The furnace was constructed for tilting the molten metal by moving the furnace using hydraulic system. The water cooling system and cooling tower were prepared to cool off the equipment with high temperature during melting is cooled off. The above process was specified in the operating procedure developed for this melting furnace system, and the operator shall operate and inspect according to the prescribed procedures. The radioactivity concentration in the sample taken in the step of tilting shall be analyzed whether they meet clearance level for self-disposal determined and publicly announced by the Commission. We can conduct self-disposal for the product of melting furnace system confirmed by the Commission as having the radioactivity concentration by nuclide not exceeding the value determined by the Commission.

Keywords: Induction melting, Melting furnace system, Decontamination, Radioactive metal waste

Revalidation of Safety Shutdown Earthquake Criteria for the Combustible Radioactive Waste Treatment Facility (CRWTF) Site

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In light of recent significant seismic events in Korea and worldwide, there is an urgent need to reevaluate the adequacy of seismic assessments conducted during facility construction. This study reexamines the ongoing viability of the Safety Shutdown Earthquake (SSE) criteria assessment for the Combustible Radioactive Waste Treatment Facility (CRWTF) site at the Korea Atomic Energy Research Institute (KAERI), originally established in 1994.

To validate the SSE assessment, we delineated 13 seismic structure zones within the Korean Peninsula and employed two distinct methodologies. Initially, we updated earthquake occurrence data from 1994 to the present year (2023) to assess changes in the site's horizontal maximum earthquake acceleration (g). Subsequently, we conducted a comparative analysis using the same dataset, contrasting the outcomes derived from the existing distance attenuation equation with those from the most recent attenuation equations to evaluate the reliability of the applied attenuation model.

The Safety Shutdown Earthquake (SSE) criterion of 0.2 g remains unexceeded, even when considering recent earthquake events since the original evaluation in 1994. Furthermore, when applying various assessment equations developed subsequently, the maximum value obtained from the previously utilized 'Donvan and Bornstein' attenuation equation is 0.1496 g, closely resembling the outcome derived from the recently employed 'Lee' reduction equation of 0.1451 g.

The SSE criteria for CRWTF remain valid in the current context, even in light of recent seismic occurrences such as the 2016 Gyeongju earthquake. Additionally, the attenuation equation employed in the evaluation consistently yields conservative results when compared to methodologies used in recent assessments. Consequently, the existing SSE criteria remain valid at present. This study is expected to serve as a valuable reference for confirming the SSE criterion assessment of similarly constructed facilities within KAERI.

Acknowledgements

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Keywords: Combustible radioactive waste treatment facility, Site characteristic evaluation, Safety shutdown earthquake evaluation

An Overseas Status Survey on Commercial Plasma Torch Melting Facilities

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A disposal of radioactive wastes is one of the urgent issues in worldwide. Considering upcoming plans for decommissioning of nuclear power plants, this problem is unavoidable and should be discussed very thoughtfully before long. There are variety of methods to deal with radioactive wastes, including Incineration process, conventional gasification process and plasma gasification process and so on. Among them, plasma gasification process is in the limelight due to its eco-friendly features and very large volume reduction effects. So, lots of countries like Japan, Taiwan, Russia, Bulgaria are already utilizing commercial plasma melting facilities and researching their own characteristics & disposal abilities and so on. Within the scope of this paper, I would like to introduce other countries current status of plasma melting facilities, and reach the conclusion on the directions to go for realistic radioactive wastes treatment.

Keywords: Radioactive wastes treatment, Plasma gasification process, Commercial plasma melting facilities

A Study on Radwaste Melting Parameters in Kilo & Mega-Watt Class Plasma Torch Melter

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Once decommissioning begins, it is expected that large amount of radioactive wastes will be produced in a short period of time. The expected amount of radioactive wastes from Kori unit 1 NPP are approximately 80,000 drums (base on 200 L). By minimizing the amount of radioactive wastes generated through decontamination and reduction, KHNP has set the final target for the amount of radioactive wastes to be delivered to the disposal site at approximately 14,500 drums. Here, plasma torch melting technology is an essential technology for radioactive wastes treatment during nuclear power plants decommissioning and operation, because of its large volume reduction effects and the diversity of disposable wastes. KEPCO KPS was able to secure experience in operating Plasma Torch Melter (PTM) by conducting a research service for ‘development of plasma torch melting system advancement technology’ at KHNP-CRI. This study will compare kilo and Mega-Watt class PTM, largely categorized into facility configurations, operating parameters, and waste treatment. Based on this study, it would be desirable to operate PTM with approximate capacity according to the frequency and amount of waste production, and suggest volume for a kilo and Mega-watt class plasma torch in the melting furnace respectively. This plays to its strengths for both a kilo and Mega-watt class PTM.

Keywords: Plasma torch melter (PTM), Radioactive wastes treatment, Radioactive wastes disposal, Volume reduction, Radwaste

Establishment of Management System for Radioactive Waste Containings Hazardous Substances

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As the acceptance criteria for low-intermediate-level radioactive waste cave disposal facilities of Korea Radioactive Waste Agency (KORAD) were revised, the requirements for characterization of whether radioactive waste contains hazardous substances have been strengthened. In addition, As the recent the Nuclear Safety and Security Commission Notice (Regulations on Delivery of Low-Medium-Level Radioactive Waste) scheduled to be revised, the management targets and standards for hazardous substances are scheduled to be specified and detailed. Accordingly, the Korea Atomic Energy Research Institute (KAERI) needs to prepare management methods and procedures for hazardous substances. In particular, in order to characterize the chemical requirements (explosiveness, ignitability, flammability, corrosiveness, and toxicity) contained in radioactive waste, it must be proven through documents or data that each item does not contain hazardous substances, and quality assurance for the overall process must be provided. In order to identify the characteristics of radioactive waste that will continue to be generated in the future, KAERI needs to introduce a management system for hazardous substances in radioactive waste and establish a quality assurance system.

Currently, KAERI is thoroughly managing chelates (EDTA, NTA, etc.), but the detailed management procedures for hazardous substances related to chemical requirements in radioactive waste in the radiation management area specified above are insufficient.

The KAERI's Laboratory Safety Information Network has a total periodic regulatory review system in place for the purchase, movement, and disposal of chemical substances for each facility. However, there is no documents or data to prove that the hazardous substances held in the facility are not included in the radioactive waste, and there are no procedures for managing hazardous substances. Therefore, it is necessary to establish procedures for the management of hazardous substances, and we plan to prepare management procedures for hazardous substances so that chemical substances can be managed according to the procedures at each facility during preliminary inspection before receiving radioactive waste. The procedure provides definitions of terms and types of management targets for each characteristic of the chemical requirements specified above (explosiveness, ignition, flammability, corrosiveness, and toxicity). In addition, procedure also contains treatment methods of radioactive waste generated by using hazardous substances and management methods of in/out, quantity, history of that substances, etc.

As the law is revised in the future, management will be carried out according to the relevant procedures. In this study, we aim to present the hazardous substance management procedures being established to determine whether radioactive waste contains hazardous substances in accordance with the revised the notice and strengthened acceptance criteria. Through this, we hope to contribute to improving reliability so that radioactive waste could be disposed of thoroughly and safely.

Keywords: Radioactive waste, Hazardous substances, Chemical requirements, Acceptance criteria

Mechanical Properties of WC-Based Coating Materials on Cold-Rolled Steel Surface Through HVOF Technique

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Domestic commercial low- and intermediate-level radioactive waste storage containers are manufactured using 1.2 mm thick cold-rolled steel sheets, and the outer surface is coated with a thin layer of primer of 10~36 μm . However, the outer surface of the primer of the container may be damaged due to physical friction, such as acceleration, resonance, and vibration during transportation. As a result, exposed steel surfaces undergo accelerated corrosion, reducing the overall durability of the container. The integrity of storage containers is directly related to the safety of workers. Therefore, the development of storage containers with enhanced durability is necessary.

This paper provides an analysis of mechanical properties related to the durability of WC (tungsten carbide)-based coating materials for developing low- and intermediate-level radioactive waste storage containers. Three different WC-based coating specimens with varied composition ratios were prepared using HVOF (high-velocity oxy-fuel) technique. These different specimens (namely WC-85, WC-73, and WC-66) were uniformly deposited on cold-rolled steel surfaces ensuring a constant thickness of 250 μm . In this work, the mechanical properties of the three different WC-based coating materials evaluated from the viewpoints of microstructure, hardness, adhesion force between substrate and coating material, and wear resistance.

The cross-sectional SEM-EDS (Scanning Electron Microscope-Energy Dispersive X-ray Spectroscopy) images revealed that elements W (tungsten), C (carbon), Ni (nickel), and Cr (chromium) were uniformly distributed within the each coating layers which was approximately 250 μm thick. The average hardness values of HWC-85 and HWC-73 were found to be 1,091 Hv (Vickers Hardness) and 1,083 Hv, respectively, while the HWC-66 exhibited relatively lower hardness value of 883 Hv. This indicates that a higher WC content results in increased hardness. Adhesion force between and substrates and coating materials exceeded 60 MPa for all specimens, however, there were no significant differences observed based on the tungsten carbide content.

Furthermore, a taber-type abrasion tester was used for conducting abrasion resistance tests under specific conditions including an H-18 load weight at 1,000 g with rotational speed set at 60 RPM. The abrasion resistance tests were performed under ambient temperatures (RT: $23\pm 2^\circ\text{C}$) as well as relative humidity levels (RH: $50\pm 10\%$). Currently, the ongoing abrasion resistance tests will include some results in this study.

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Keywords: Radioactive waste storage container, WC-based coating, HVOF, Hardness, Adhesion test, Abrasion tests

Operation of Radioactive Waste Management System

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KAERI has developed a Radioactive Waste Information Management System (RAWINGS) to manage the life-cycle information from the generation to the disposal of radioactive waste, in compliance with the low- and medium-level radioactive waste acceptance criteria (WAC).

In the radioactive waste management process, the preceding steps are to receive waste history from the waste generators. This includes an application for a specified container with a QR label, pre-inspection, and management request. Next, the succeeding steps consist of repackaging, treatment, characterization, and evaluating the suitability of disposal, for a process to transparently manage radioactive wastes.

Since the system operated in 2021, The system is enhanced to manage dynamic information, including the tracking of the location of radioactive waste and the repackaging process. Small packages of waste could be classified as either radioactive or clearance waste during pre-inspection. Furthermore, waste generated in the past has already been packaged in drums, and a new algorithm has been developed to apply the repackaging when reclassification is required.

All radioactive waste with the unique ID number on the specific container is managed within a database, the total amount and history of waste are managed, and statistical information is provided. This system is continuously be operated and developed to oversee life-cycle information, and serve as the foundational database for the Waste Certification Program (WCP).

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Keywords: Radioactive waste, WAC, QR code, Database, Specific container

A Preliminary Result on Comparison of Concentration Distribution by Nuclide in Concrete and Soil Wastes From KRR 1&2

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In the decommissioning site of Korean Research Reactor 1&2 (KRR-1&2), according to Low and Intermediate-level Radioactive Waste Disposal Acceptance Criteria of the Korea Radioactive Waste Agency (WAC-SIL-2022-1), characteristics of radioactive waste was conducted on approximately 550 drums of concrete and soil waste for a year starting from 2021. Among them, 50 drums of concrete waste transported and disposed to Gyeongju LILW disposal facility at the end of 2022. For the remaining approximately 500 drums of concrete and soil waste stored on-site, they were reclassified into two categories: permanent disposal grade and clearance grade. This classification was based on calculating the sum of fractions (SOF) per drum for each radionuclides. The plan is to dispose of around 200 drums in the permanent disposal grade and about 300 drums in the clearance grade by the end of 2023.

Since concrete and soil decommissioning wastes are generated in large quantities over a short period with similar origins, they were grouped within five drums as suggested by the acceptance criteria. Mixed samples were collected from each group and used for radionuclide analysis. When utilizing mixed samples, three distinct samples are collected and analyzed for each group. The maximum value among these three radionuclide analysis results is then uniformly applied as the radionuclide concentration value for all drums within that group. Radioactive nuclides contained in similar types of radioactive waste with similar origins can be expected to have some statistical distribution. However, There has been no verification as to whether the maximum value among the three mixed samples exists within the statistical distribution or if it deviates from this distribution to represent a different value.

In this study, we confirmed characteristics of radionuclide concentration distribution by examining and comparing radionuclide concentration distributions for radioactive wastes drum grouped for nuclear characteristic among 50 concrete wastes drum disposed in year 2022 and 500 concretes & soils drum scheduled for disposal (clearance or permanent disposal) in year 2023. In particular, when comparing tritium to other nuclides, it was observed that the standard deviation for the distribution of maximum values was approximately 318 times larger.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (KAERI) [Grant No.521240-23, South Korea].

Keywords: Clearance, Radionuclides analysis, Decommissioning, Concrete and soil waste, KRR1&2

Review for Disposal Suitability of Drums Compressed With Super-High Pressure

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Radioactive waste (hereinafter referred to as mixed waste) containing hazardous substances (heavy metals, organic and inorganic waste liquids, asbestos, etc.) has been continuously generated from domestic nuclear power plants, nuclear facilities, and other industrial facilities, and heavy metals were released during the dismantlement of Kori Unit 1 and Wolsong Unit 1. Lead, cadmium, mercury, arsenic, asbestos, decontamination waste liquid (organic/inorganic waste liquid), etc. may be generated.

Although hazardous waste related to the nuclear industry continues to be generated, only the regulation direction for hazardous substances is presented in the provisions related to hazardous substances in the delivery regulations for low and intermediate-level radioactive waste and the acceptance criteria for low and intermediate-level radioactive waste disposal facilities. In particular, because there is no clear definition of “hazardousness” and specific standards such as concentration and characteristics for classification of hazardous substances, as well as hazard removal procedures when the hazardousness of radioactive waste is confirmed, no hazardous substances have been delivered in Korea to date and many mixed wastes are stored at each generation facility or at the NPP.

As a plan to improve delivery standards related to mixed waste is being prepared recently, it is believed that if the acceptance standards are revised accordingly, it will be possible to confirm the suitability for disposal of drums produced after the establishment of the acceptance standards in 2015. However, it is believed that securing disposal suitability for waste that was packed in 200L drums and compressed under super high pressure in the absence of specific technical standards and regulatory guidelines for the disposal of radioactive waste containing hazardous substances would still remain a difficult problem.

In this report overseas acceptance standards related to hazardous waste were reviewed and a plan to secure the disposal suitability of 200 L drums compressed with of super high pressure was proposed.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. RS-2023-00233621).

Keywords: Compression, Supper, Nuclide analysis, Hazardous, Mixed waste

Regulatory Clearance of Organic Liquid Radioactive Waste by Landfill After Incineration at KAERI

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The nuclear facilities at Korea Atomic Energy Research Institute (KAERI) have generated a variety of organic liquid radwaste and radiation levels are also varied. At KAERI, the organic liquid radwaste has been stored at Radioactive Waste Treatment Facility (RWTF) temporarily due to the absence of the recognized treatment technique while inorganic liquid radwaste can be treated by evaporation, bituminization, and solar evaporation process. The organic liquid radioactive waste such as spent oil, cutting oil, acetone, ethanol, etc. was generated from the nuclear facilities at KAERI. Among the organic liquid radioactive wastes, spent oil is particularly significant.

According to the nuclear safety act, radioactive waste can be cleared by incineration and landfilling if it meets the criteria of less than 10 $\mu\text{Sv/h}$ for individual dose and 1 person – Sv/y for collective dose. Dose assessment was performed on some organic liquid radioactive waste with a very low possibility of radioactive contamination stored in RWTF at KAERI. As a result, it was confirmed that some wastes met the regulatory clearance standards.

Based on this, it was approved by the regulatory body, and this became the first case in Korea and KAERI for permission for regulatory clearance of organic liquid radioactive waste by landfill after incineration.

Acknowledgements

This work was partly supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (RS-2023-00242179, Development of Integrated Processing Technology for High and Low Concentration Liquid Radioactive Waste Generated during Decommissioning of NPP) and This work was supported by a research grant from Korea Atomic Energy Research Institute (Grant No. 521210-23).

Keywords: Organic liquid radioactive waste, Regulatory clearance, Incineration, Landfill

Development of Cementation Process for Radioactive Sludges

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Wet solid wastes including spent ion exchange resins, evaporator concentrates and sludges require solidification to transform wastes into an acceptable solid, monolithic form for final disposal. The development of the process control program for the solidification of radioactive sludges generated at nuclear power plants has been in progress to provide reasonable assurance that the solidified product will meet the established waste acceptance criteria for solidified waste. A mobile solidification system to produce the solidified waste in the size of a 200 L drum was used, which adopts the in-line mixing method where the waste and binder are mixed and then transferred to the disposable container. To simulate radioactive sludges, non-radioactive sludges are synthesized and the specimens are prepared by using them. The qualification tests on the prepared specimens including the compressive strength test, the thermal cycling test, the irradiation test, the leach test, the immersion test, etc. have been performed to qualify recipes for a range of waste compositions. The results of the tests will be analyzed and discussed.

Keywords: Solidification, Stabilization, Cementation, Immobilization, Radioactive wastes, Sludges

Investigation on the Characteristics of Spent Scintillation Liquid

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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, it is required to investigate the nature of the waste such as specific radioactivity and moisture content. The analysis results of spent scintillation liquid generated at Wolsong nuclear power plants will be discussed. An overview of the technical approaches available for the treatment of organic radioactive waste will be additionally provided.

Keywords: Scintillation liquid, Scintillation cocktail, Organic radioactive wastes, Liquid scintillation counter, Liquid scintillator

~~Preliminary Study on the Sealing Factor Evaluation Method Considering Measurement Uncertainty for Radioactive Waste~~

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Withdrawal

A Study on Basic Characterization of Inner Radioactive Waste in Large Concrete Drum

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In the Kori power plant radioactive waste storage, the concentrated waste and spent resin drums generated in the past are repacked and stored in large concrete drums. Four 200 L drums of solidified concentrated waste are packed in the square concrete. One 200 L drum of spent resin is packed inside the round concrete. In order to build a foundation for disposal of large concrete drums that generated in the past, it is necessary to develop a large concrete drum handling device and disposal suitability evaluation technology.

In order to build handling equipment and establishment of disposal base, such as weight and volume, of square and round concrete containers must be identified. In addition, waste information, such as the production record of the built in drum and the type of contents, is required.

Therefore, this study plans to comprehensively review the characteristics of the waste by investigating the structure of square and round concrete containers and the records of internal drum production.

Keywords: Large concrete drum, Radioactive waste

Review of Geomelt Vittrification

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The treatment of solid radioactive waste can be divided into Mechanical (compaction), Thermal (Plasma), Melting (metal), Chemical (e.g. acid digestion) and Biochemical (e.g. bacteria). Among them, industrial thermal technologies include geomelt, Vittrificaion, Hip Ceramic, Incinerator, Pyrolysis, Plasma and Melting.

In this study, the characteristics, status and advantages of geomelt vittrification were reviewed.

Vittrification has long been considered an ideal choice for high-level radioactive waste by regulators internationally, because of its expected durability over hundreds of thousands of years.

Geomelt vittrification is a highly flexible technology for hazardous and radioactive waste treatment. Uses electricity to melt waste materials to either destroy or immobilize contaminants. Final product is identical to natural obsidian very durable and resistant to weathering Geomelt vittrification creates ultra stable glass that is typically 10 times stronger than concrete, and more durable than granite or marble. Its leach resistance is among the highest of all materials in the world. In addition, contaminated soil, sludge, metals, organic matter, and bulky D&D debris can be treated simultaneously without pretreatment steps such as size reduction and sorting. Geomelt vittrification can be deployed in variety of in ground, in container or hybrid in cell treatment.

Geomelt vittrification have been treating radioactive waste and hazardous waste since the 1990s, treatment in the U.S., UK, Australia, Japan and other countries. Initially developed by Pacific Northwest National Laboratory in the U.S., GeoMelt vittrification has been used successfully around the world for the U.S. Department of Energy (DOE) in Hanford and at Sellafield in the UK.

Keywords: Vittrification, Geo melt

Study on Characteristics of Contaminated Ascon in Gyeongju LILW Repository in Terms of Waste Acceptance Criteria

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KORAD (Korea Radioactive Waste Agency, <http://www.korad.or.kr>) has stored slightly contaminated ascon (asphalt coated concrete mixture) that was introduced to Gyeongju repository about a decade ago waiting for a final disposal. It is believed to be mainly contaminated by radioisotope ¹³⁷Cs due to impurities introduced from the outside during the ascon manufacturing process.

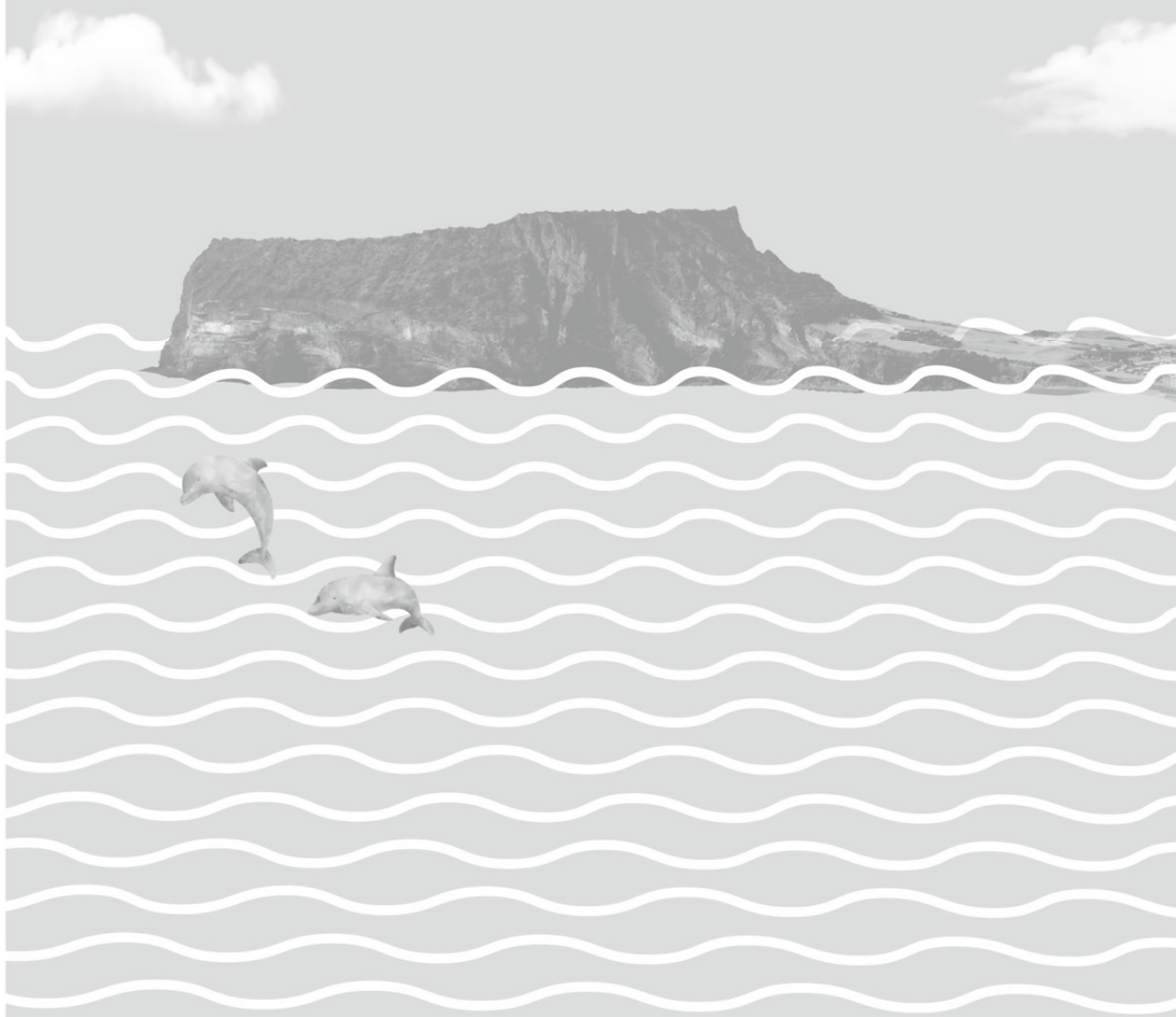
We studied characteristics of the radioactive waste to see whether this material would be proper enough to be disposed in Gyeongju LILW repository or be other ways to reduce the disposal volume including self-disposal before its final disposal otherwise.

KORAD looked into the properness of characteristics of ascon in terms of WAC (Waste Acceptance Criteria) documented by KORAD that includes general chemical and physical properties of asphalt, density, size of grains, content of organic material and possibility of existence of chelate materials that qualitatively limited to be disposed by the criteria. And other associated characteristics such as gas generation and bio degradation were also investigated. Based on the data obtained from the study, we proposed various plausible solutions in associated with operational and disposal safety and economic view points. This study will be used for KORAD's decision on how to control and safely dispose the spent ascon within a reasonable time period. And also those experiences may be applied for other LILW issues that require treatment or conditioning of radioactive wastes in the future.

Keywords: Spent ascon, Treatment, WAC (waste acceptance criteria), Disposal, Waste characteristics

5분과

제염해체 (ORAL)



Uranium-Incorporated Iron (Oxyhydr) Oxide in Geodisposal Systems: The Effects of Biogeochemical Perturbations on Long-Term Stability

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In all geodisposal scenarios it is key to understand the interaction of radionuclides with mineral particles during their formation/recrystallisation. Studying processes at the molecular scale provides insight into long-term radionuclide behaviour. Uranium is a significant radionuclide in higher activity wastes destined for geological disposal, and iron (oxyhydr) oxides (e.g. goethite, α -FeOOH) are ubiquitous in and around these systems, formed via processes including metal corrosion and microbially induced reactions. There are numerous reports of uranium-incorporation into iron (oxyhydr) oxides, therefore it has been suggested that they may be a barrier to uranium migration in geodisposal systems. However, long-term stability of these phases during environmental perturbations are unexplored. Specifically, U-incorporated iron (oxyhydr) oxide phases may interact with Fe(II) and sulphide from biological or geological origin. Firstly, electron transfer occurs between adsorbed Fe(II) and iron oxyhydroxides, with potential for changes in the speciation of incorporated uranium e.g. oxidation state changes and/or release. Secondly, on exposure to aqueous sulfide, iron (oxyhydr) oxides undergo reductive dissolution and recrystallisation to iron sulphides. Understanding the fate of incorporated uranium during these process is key to understanding its long term behaviour in subsurface systems.

A series of experimental studies were undertaken where U(VI)-goethite was synthesized then reacted with either aqueous Fe(II) or S(-II), and the system monitored over time using geochemical analysis and X-ray absorption spectroscopy (XAS) techniques e.g. U L_{III}-edge and MIV-edge HERFD-XANES. Reaction with aqueous Fe(II) resulted in electron transfer between Fe(II) and U(VI)-goethite, with > 50% U(VI) reduced to U(V). XAS analysis revealed that U remained within the goethite structure, and electron transfer only occurred within the outermost atomic layers of goethite. which led to U reduction. Rapid reductive dissolution of U(VI)-goethite occurred on reaction with sulfide at pH7. A transient release of aqueous U was observed during the first day, likely due to uranyl(VI)-persulfide species. However, U was retained in the solid phase in the longer term. In contrast, the sulfidation of U adsorbed to ferrihydrite at pH 12.2 led to the immediate release of U (< 10% U_{total}) associated with a colloidal erdite (NaFeS₂·2H₂O) phase. Moreover, in the bulk phase the surface of ferrihydrite was passivated by sulfide, and U was found to have been trapped within surface associated erdite-like fibres.

Overall, these studies further understanding of the long-term behaviour of U-incorporated iron (oxyhydr)oxides supporting the overarching concept of iron (oxyhydr) oxides acting as a barrier to U migration.

Keywords: Uranium, Geological disposal, X-ray absorption spectroscopy, Radioactive waste

Recent Advances in Geopolymer Waste Forms to Immobilize Radioactive Wastes

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The immobilization of low- and intermediate-level radioactive waste (LILW) is crucial for its final disposal in repositories. While cementitious waste forms have conventionally been used for immobilizing various LILWs, they suffer from several issues, including poor durability, low resistance to leaching, and limited waste loading capacity. As an alternative, alkali or acid-activated geopolymer waste forms have garnered global attention. Unlike cementitious waste forms, geopolymer waste forms exhibit excellent physicochemical characteristics due to their three-dimensional amorphous structure and low calcium content. In this work, we provide an overview of geopolymer waste form research being conducted in countries such as Japan, the United Kingdom, the European Union, and South Korea. We specifically focus on the immobilization of soil waste, spent ion exchange resins, organic liquid waste, and evaporator concentrate (borate waste). We also identify the factors influencing the physicochemical characteristics of geopolymer waste forms and their immobilization performance. We propose a guide for optimizing the molar mixing formulations of geopolymer waste forms, including the selection of appropriate precursor materials. Additionally, we discuss the future prospects and significant challenges in the field of geopolymer waste forms that need to be addressed for their application in radioactive waste management.

Acknowledgements

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Keywords: Geopolymer, Waste form, Waste acceptance criteria, Radioactive waste

A Review on the Linkage of Input Data for the Development of FDP at the Decommissioning Nuclear Power Plant

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The decommissioning of domestic Nuclear Power Plants (NPPs) in Korea is expected to begin with the Kori-1, which was permanently shutdown in 2017. In addition, Wolsong-1 has been also permanently shutdown, and another type will be the decommissioning project following Kori-1. KHNP is promoting operation and decommissioning projects as the owner of NPPs, and the Central Research Institute (CRI) has been developing a Final Decommissioning Plan (FDP) for the decommissioning license document. The FDP consists of 11 major chapters in the order of overview of the project, characteristic evaluation, safety assessment, radiation protection, decontamination & dismantlement activities, waste management, etc. The contents described in each chapter are individual chapters, but there are also parts that consider the connection with other chapters. The CRI, which develops the FDP for the first decommissioning project in Korea, has spent a lot of time and effort considering this and has been proceeding through trial and error until the present stage. Therefore, this study aims to explain the current status of FDP, a license document for domestic decommissioning projects, and the link between major input data in major chapters.

It can be said that System, Structure, and Components (SSCs) subject to dismantling are considered as the scope of FDP. Chapters that perform estimations on these dismantling targets may include safety assessments, exposure dose assessments for workers and residents, and waste inventory assessments. Therefore, an important part of performing the estimation works is to consider the entire scope of decommissioning activities, and as a way, it can start from data based on the inventory data. After generating the inventory data, the waste treatment classification for the inventory is designated by reflecting the results of the characterization. In addition, for cost estimation, the cost of decommissioning project is predicted by inputting some data (i.e., UCF) such as work process, number of workers, and time required for each item with data reflected in quantity and characterization.

After that, based on these inventory, characterization, and UCF data, accident scenarios and industrial safety evaluation are performed for the safety assessment. The worker exposure dose is estimated by considering the dose rate of the workspace with these data. In the case of the amount of waste, the final amount of waste is estimated by considering the factors of reduction and decontamination. In summary, the main estimation contents of FDP are evaluated by adding elements required for the purpose of each chapter from data combined with inventory, characterization, and UCF, so the contents of these chapters are based on the logic of considering the entire scope of decommissioning in common.

Keywords: Decommissioning, Final decommissioning plan, Decontamination and dismantlement

Predicting the Distribution Coefficients of Radionuclides Using Machine Learning

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Nuclear facilities present the important task related to the migration and retention of radioactive contaminants such as cesium (Cs), strontium (Sr), and cobalt (Co) for unexpected events in various environmental conditions. The distribution coefficient (K_d) is important factor for understanding these contaminants mobility, influenced by environmental variables. This study focusses the prediction of K_d values for radionuclides within solid phase groups through the application of machine-learning models trained on experimental data and open source data from Japan atomic energy agency. Three machine-learning models, such as the convolutional neural network, artificial neural network, and random forest, were trained for prediction model of the distribution coefficient (K_d). Fourteen input variables drawn from the database and experimental data, including parameters such as initial concentration, solid-phase characteristics, and solution conditions, served as the basis for model training. To enhance model performance, these variables underwent preprocessing steps involving normalization and log transformation. The performances of the models were evaluated using the coefficient of determination. These results showed that the environmental media, initial radionuclide concentration, solid phase properties, and solution conditions were significant variables for K_d prediction. These models accurately predict K_d values for different environmental conditions and can assess the environmental risk by analyzing the behavior of radionuclides in solid phase groups. The results of this study can improve safety analyses and long-term risk assessments related to waste disposal and prevent potential hazards and sources of contamination in the surrounding environment.

Acknowledgements

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Keywords: Radionuclides, Clay mineral, Distribution coefficient, Prediction, Machine learning

Kinetics of Thermal Decomposition Reactions and Changes in Structures During the Course of Carbonization of Strong Acidic Cation Exchange Resin

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The process of carbonization followed by a high-temperature halogenation removal of radionuclides is a promising approach to convert low-radioactivity spent ion-exchange (IE) resins into free-releasable non-radioactive waste. The first step of this process is to convert spent ion-exchange resins into the carbon granules that are stable under high-temperature and corrosive-gas flowing conditions.

This study investigated the kinetics of carbonization of cation exchange resin (CER) and the changes in structures during the course of carbonization to 1,273 K. Both of model-free and model-fitted kinetic analysis of mixed reactions occurring during the course of carbonization were first conducted based on the non-isothermal TGAs and TGA-FTIR analysis of CER to 1,272 K. The structural changes during the course of carbonization were investigated using the high-resolution FTIR and C-13 NMR of CER samples pyrolyzed to the peak temperature of each reaction steps established by the kinetic analysis.

Four individual reaction steps were identified during the course of carbonization to 1,273 K. The first and the third steps were identified as the dehydration and the dissociation of the functional group of $\text{—SO}_3\text{H}^+$ into SO_2 and H_2O , respectively. The second and the fourth steps were identified as the cleavage of styrene divinyl benzene copolymer and carbonization of pyrolysis product after the cleavage, respectively.

The temperature and time positions of the peaks in the DTG plot are nearly identical to those of the peaks of the Gram Schmidt intensity of FTIR. The structural changes in carbonization identified by high-resolution FTIR and DTG are in agreement with those by C-13 NMR. The results of a detailed examination of the structural changes according to NMR and FTIR were in agreement with the pyrolysis gas evolution characteristics as examined by TGA-FTIR.

Keywords: Kinetic analysis, Cation exchange resin, Thermal decomposition reaction, Carbonization, Structural change, Gas-evolution characteristics

Proton Exchange Membrane Water Electrolyzer for Tritium and Hydrogen Isotope Separation

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Hydrogen isotope separation involves the separation of hydrogen, deuterium, tritium, and their isotopologues. It is an essential technology for removing radioactive tritium contamination and for obtaining valuable hydrogen isotope resources. Among various hydrogen isotope separation technologies, water electrolysis technology exhibits a high separation factor. Consequently, the electrolysis of tritiated water is of paramount importance as a tritium enrichment method for treating tritium-contaminated water and for analyzing tritium in environmental samples.

More recently, hydroelectrolysis technology, which utilizes proton exchange membranes (PEM) to reduce water inventory, has gained favor over traditional alkaline hydroelectrolysis. Nevertheless, it is crucial to decrease the hydrogen permeability of the PEM in order to mitigate the explosion risk associated with tritium hydrogen electrolysis devices. Additionally, efforts are needed to enhance the hydrogen isotope selectivity of the PEM and optimize the manufacturing process of the membrane-electrode assembly (MEA), thereby improving both hydrogen isotope separation performance and water electrolysis efficiency.

In this presentation, we will delve into two key aspects. Firstly, we'll explore the reduction of hydrogen permeability and the enhancement of the hydrogen isotope separation factor in PEM through the incorporation of 2D nanomaterial additives. Secondly, we'll examine the influence of various MEAs preparation methods on electrolysis and isotope separation performances. Lastly, we will discuss the effectiveness of the developed system in separating deuterium and tritium.

Acknowledgements

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Keywords: Tritium, Isotope separation, Electrolysis, Proton exchange membrane

Manufacturing of Mobile High Intensity Focused Ultrasound Equipment for Decontamination of Metallic Radioactive Waste and Demonstration in NPP

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Despite its advantages such as safety, unnecessary pretreatment, and decontamination of waste with complex geometry, conventional ultrasonic decontamination technology has been only used to remove loose contaminants, oil and grease, not fixed contaminants due to the limitations in increasing the intensity in the high frequency range. Thus, ultrasound has been used as an auxiliary method to accelerate chemical decontamination of radioactive wastes or chemicals were added to the solution to increase the decontamination efficiency.

The recently developed high-intensity focused ultrasound (HIFU) decontamination technology overcomes these limitations by combining multiple frequencies of ultrasonic waves in a specific arrangement, making it possible to remove most fixed contaminants, including radioactive micro particles less than 1 micrometer within half an hour.

KEPCO NF and EnesG developed mobile HIFU decontamination equipment and successfully demonstrated the decontamination effect on various radionuclides found in nuclear power plants by treating radioactive metal waste to the level below free release criteria.

The mobile HIFU decontamination equipment used in the demonstration can be operated anywhere where water is supplied, including controlled area in nuclear power plants, and is expected to be used widely for decontamination and free release of metal radioactive wastes.

Keywords: High intensity focused ultrasound, Decontamination, Solid radioactive wastes, Low level waste, Decommissioning

Certified Reference Materials for Decommissioning Nuclear Power Plants

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The Korea Research Institute of Standards and Science has developed certified reference materials (concrete, soil, and metal radioactive liquid) for measuring gamma-emitting radionuclides to improve and maintain the quality assurance and quality control of the radioactivity measurement in decommissioning nuclear power plants. The raw materials that make up each CRM were mixed in an appropriate ratio with radionuclides. For certification and homogeneity assessment, 10 bottles were randomly selected, two sub-samples were collected from each bottle, and radionuclides were measured via HPGe gamma spectrometry. The results of the homogeneity tests using a one-way analysis of variance on the radionuclides in the CRMs fulfilled the requirements of ISO Guide 35. Coincidence summing and self-absorption correction were performed on measurement results by introducing the Monte Carlo efficiency transfer code and Monte Carlo N-Particle transport code. In concrete analysis, the reference values for five radionuclides (^{60}Co , ^{241}Am , ^{134}Cs , and ^{137}Cs) in the CRM were in the range of 15-40 Bq/kg, and the expanded uncertainty was within 10% ($k = 2$). In soil analysis, the reference values for the ^{137}Cs and ^{60}Co were 118.7 and 124.4 Bq/kg, and the expanded uncertainty was within 10% ($k = 2$). In metal radioactive liquid analysis, the reference values for ^{134}Cs , ^{137}Cs and ^{60}Co in the CRM were in the range of 200-270 Bq/kg, and the expanded uncertainty was within 7% ($k = 2$).

Keywords: Nuclear power plants, Decommissioning, Radioactivity measurement, Certified reference material, Concrete, Soil, Metal

Performance Evaluation of Large-Capacity Decontamination System for Radioactively Contaminated Soil Using Water-Washing Treatment Technology

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Economical radioactive soil treatment technology is essential to safely and efficiently treat of high-concentration radioactive areas and contaminated sites during operation of nuclear power plants at home and abroad. This study is to determine the performance of BERAD (Beautiful Environmental construction's RAdioactive soil Decontamination system) before applying magnetic nanoparticles and adsorbents developed by the KAERI (Korea Atomic Energy Research Institute) which will be used in the national funded project to a large-capacity radioactive soil decontamination system.

BERAD uses Soil Washing Process by US EPA (402-R-007-004 (2007)) and can decontaminate 0.5 tons of radioactive soil per hour through water washing and/or chemical washing with particle size separation. When contaminated soil is input to BERAD, the soil is selected and washed, and after going through a rinse stage and particle size separation stage, it discharges decontaminated soil separated by sludge of less than 0.075 mm.

In this experiment, the concentrations of four general isotopes (A, B, C, and D which are important radioisotopes when soil is contaminated by them.) were analyzed by using ICP-MS to compare before and after decontamination by BERAD. Since BERAD is the commercial-scale pilot system that decontaminates relatively large amount of soil, so it is difficult to test using radioactive isotopes. So important general elements such as A, B, C, and D in soil were analyzed. In the study, BERAD decontaminated soil by using water washing. And the particle size of soil was divided into a total of six particle size sections with five sieves: 4 mm, 2 mm, 0.850 mm, 0.212 mm, and 0.075 mm.

Concentrations of A, B, C, and D in the soil particles larger than 4 mm are almost the lowest regardless of before and after decontamination by BERAD. For soil particles less than 4 mm, the concentrations of C and D decreased constantly after BERAD decontamination. On the other hand, the decontamination efficiency of A and B decreased as the soil particle became smaller, but the concentrations of A and B increased for the soil particle below 0.075 mm. As a result, decontamination efficiency of one cycle using BERAD for all nuclides in soil particles between 4 mm and 0.075 mm is about 45% to 65 %.

Acknowledgements

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Keywords: Radioactively contaminated soil, Soil water-washing, Commercial scale, Soil decontamination efficiency

A Study on Treatment of Historic Waste (Focusing on Large Concrete Drum)

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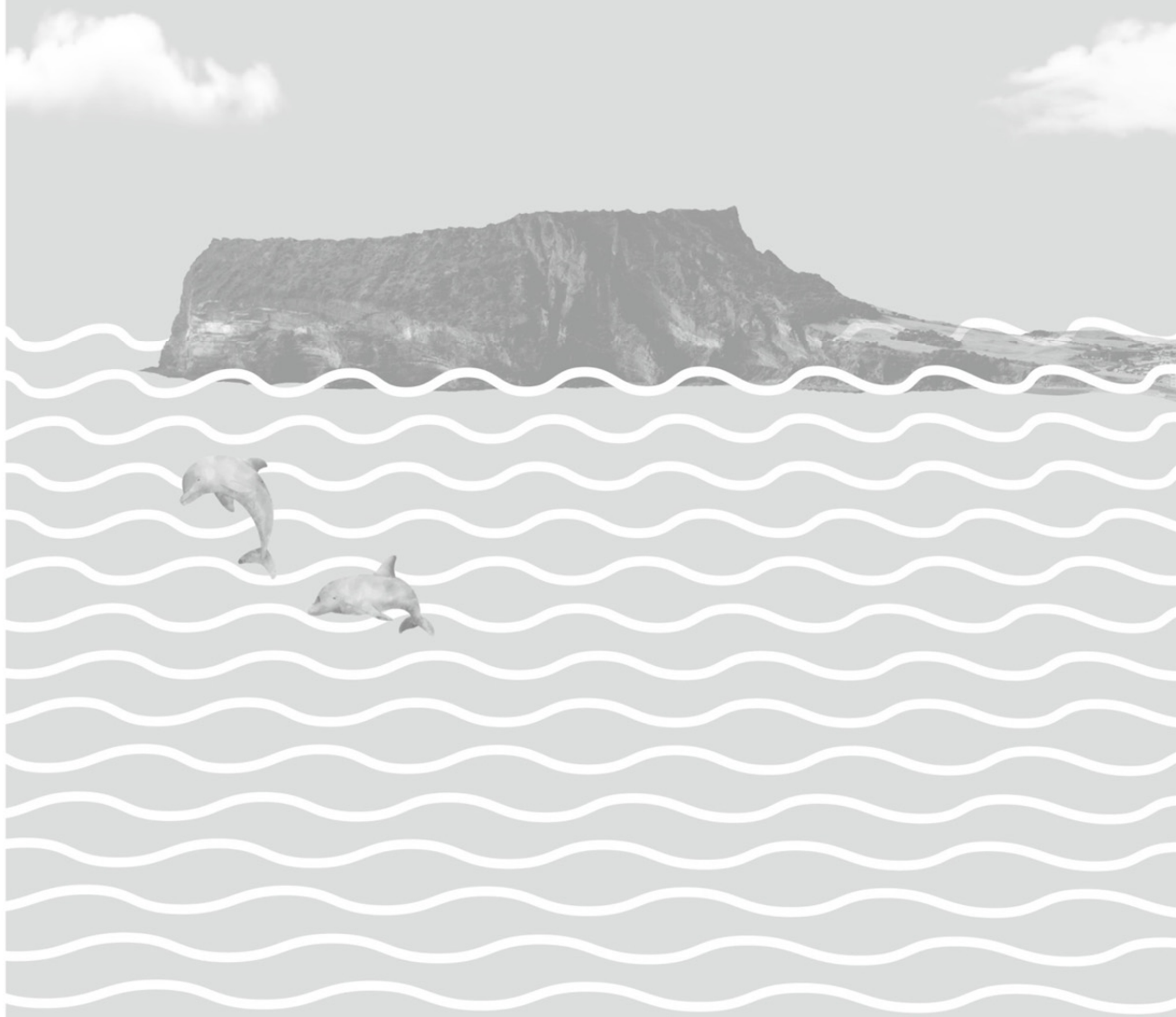
During the operation of nuclear power plant (NPP), the concentrates and spent resin are generated. They show relatively high radioactivity compared to other radioactive waste, such as dry active waste, charcoals, and concrete wastes. The waste acceptance criteria (WAC) of disposal facility defines the structure and property of treated waste. The concentrates and spent resin should be solidified or packaged in high integrity container (HIC) to satisfy the WAC in Korea. The Kori NPP has stored history waste. The large concrete package with solidified concentrates and spent resin. The WAC requires identification of 18 properties for the radioactive waste. Since some of the properties are not clearly identified, the large concrete packages could not satisfy the WAC in this moment.

The generation of the large concrete package (rectangular type and cylindrical type), pre-treatment of the package, treatment of inner drum, process development for clearance waste, etc. will be discussed in this paper. In addition, the conceptual design of whole treatment process will be discussed.

Keywords: Large concrete package, Segmentation, radioactive waste, Heat treatment, Decommissioning

5분과

제염해체 (POSTER)



Proposal of Treatment Process to Secure Disposal Safety for Permanent Disposal of Aluminum Radioactive Waste

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When aluminum is in an alkaline state, the aluminum oxide film surrounding aluminum is dissolved and moisture penetrates the exposed aluminum surface, causing corrosion of aluminum. At this time, hydrogen gas is generated and there is a risk of explosion due to the generated hydrogen gas.

Aluminum radioactive waste is difficult to permanently dispose of because it does not meet the standards for the acquisition of low- and intermediate-level radioactive waste cave disposal facilities currently managed and operated by the Korea Nuclear Environment Corporation. However, because of this risk, it is necessary to study how to safely treat and dispose aluminum waste. In this study, overseas cases were investigated and analyzed to ensure the safety of aluminum waste disposal, and the current status of aluminum radioactive waste generated during decommissioning of the Korea Research Reactor 1&2 and a treatment plan to secure disposal suitability were presented. The process of removing a little remaining oxygen in molten steel during the reduction of iron oxide in the iron refining process is called deoxidation, and a representative material used for deoxidation is aluminum. In the case of metal melting decontamination, which is one of the decontamination processes of radioactive metal waste, a method of treating aluminum waste by using aluminum as a deoxidizer is proposed.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (KAERI) [Grant No. 521240-23, South Korea].

Keywords: Aluminum, Radioactive waste, Disposal, Melting, Deoxidizer

Activated RVI Evaluation by Using Surveillance Capsule in Decommissioning NPP

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In order to evaluate the integrity of the reactor pressure vessel, various test specimens necessary to identify irradiation embrittlement. The degree of irradiation embrittlement of the vessel material by neutrons, from the construction to the end of the life of the plant, is evaluated by a monitoring plan that called surveillance program (a series of all plans to analyze and evaluate embrittlement through various tests and analyzes by placing a test piece inside the reactor pressure vessel and taking out a piece at an appropriate time according to the number of operation years and taking necessary measures for safe operation).

The reactor monitoring specimens for Kori Unit-1 are located by axis at S (57°), T (67°), R (77°), N (237°), P (247°) and V (257°). Six surveillance capsules are attached to the inside of the pressure vessel around the core and to the outside of the thermal shield. This surveillance container determines the withdrawal timing of the surveillance container according to the provisions of ASTM E185-82.

In the monitoring test piece, there are neutron dosimeter materials to measure and evaluate the irradiated neutron flux, and Ni, Cu, Fe, Co-Al, Cd, and shielded Co-Al monitors are wired in the monitoring container. Each axial position is contained in a spacer hole. The neutron dosimetry monitor measures the neutron dose using isotopes produced by neutrons during operation of the reactor.

The Al-Co specimen, which can evaluate the degree of radioactivity of cobalt, is located on the lower part of the specimen. The content of Co in the Al-Co specimen is 0.15%, and when expressed in ppm, it is 1,500 ppm, which is similar to the cobalt content of 1,414 ppm in the internal structure of the reactor vessel presented in NUREG-3474. If the radiation value of the Al-Co sample in the reactor monitoring specimen can be measured, the radiation value of the internal structure of the reactor can be indirectly compared. Since the monitoring specimen is located outside of the thermal shield, radiation should be less than that of the thermal shield. Korea Reactor Monitoring Technology performed gamma measurement on Al-Co specimens in 6 monitoring specimens, and although there are differences depending on the sample, it shows radioactivity values around the order of $1\text{E}+07$ dps/g, or Bq/g. In conclusion, it is thought that using this measurement values, it is possible to verify the evaluation of internal structure radiation for Kori unit-1 decommissioning.

Keywords: Decommissioning, Surveillance capsule, RVI, Kori

Periodic Safety Review (PSR) for Decommissioning NPP

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The periodic safety review (PSR), for all operating nuclear power plants in Korea, has been conducted in accordance with SSG-25, a guideline suggested by the IAEA. The PSR is performed through the review of the regulatory body after the operator's self-evaluation. In order to guarantee a high level of safety in consideration of the changed environment, such as operating experience (OE) and technology development, it should be comprehensively and integrately performed, and it is also carried out every 10 years after the operation permit.

However, in case that all or part of the reactor facilities have been permanently shut down, such as Kori Unit 1 and Wolsong Unit 1, Around a half of reactor facilities are not in operation. The periodic safety evaluation may not be conducted for unused parts if there is no safety hazard and if there are some difficulties for applying periodic safety evaluation.

In considering that the biggest purpose of PSR safety (by PSR definition of KINS guideline) is to improve and accumulated factors such as aging deterioration, facility change, operation experience, and technological development for operating nuclear power plants. It refers to a comprehensive safety evaluation that is periodically performed during the period of operation of a nuclear power plant. It is necessary to review whether PSR should be performed for a nuclear power plant that is permanently shut down after nuclear power plant operation is terminated. Also, in IAEA SSR 2/2 Rev1, it is defined that PSR is performed during the nuclear power plant operation period. "Requirement 12: Periodic safety review, Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant's operating lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources".

Recently, Kori Unit 1 and Wolsong Unit 1 were decided to permanently shut down in June 2017 and December 2019, and are currently being prepared for decommissioning. According to the Wolsong decommissioning plan, decontamination and demolition will be completed by 2032. The PSR for permanent shutdown of Kori Unit 1 was submitted to the regulatory body in December 2018 and is under approval review. In the case of the permanent shutdown PSR of Wolsong Unit 1, the project will be launched in May 2023 and the PSR will be submitted to the regulatory body in May 2024. In the case of Wolsong Unit 1, it is necessary to operate the various systems, including the systems related to the spent fuel storage tank, even during the period of permanent shutdown. Such as the heavy water related systems used in common with Wolsong Unit 2, are essential operating systems. Based on Basic Subject Index (BSI), 112 out of 218 systems require operation, indicating that about 50% of systems require operation even after permanent shutdown. Decommissioning of systems and equipment will begin after the transfer to modular air-cooled canister storage (MACSTOR) by the end of 2025, and then in-depth discussions will be needed whether PSR evaluation is meaningful.

Keywords: Decommissioning, PSR, Wolsong

Feasibility Testing of Integrated Heating and Grinding Equipment to Enhanced the Process for Volume Reduction of Radioactive Concrete Waste

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Decommissioning waste is generated with various types and large quantities within a short period. Concrete, a significant building material for nuclear facilities, is one of the largest decommissioning wastes, which is mixed with aggregate, sand, and cement with water by the relevant mixing ratio. Recently, the proposed treatment method for volume reduction of radioactive concrete waste was proven up to scale-up testing using unit equipment, which involved sequentially thermomechanical and chemical treatment. According to studies, the aggregate as non-radioactive material is separated from cement components with contaminated radionuclides as less than clearance criteria, so the volume of radioactive concrete waste is decreased effectively. However, some supplementation points were presented to commercialize the process. Hence, the process requires efficiency as possible to minimize the interface parts, either by integration or rearranging the equipment.

In this study, feasibility testing was performed using integrated heating and grinding equipment, to supplement the possible issue of generated powder and dust during the process. Previously, heat treatment and grinding devices were configured separately for pilot-scale testing. But some problems such as leakage and pipe blockage occurred during the transportation of generated fine powder, which caused difficulties in maintaining the equipment. For that reason, we studied to reduce the interface between the equipment by integrating and rearranging the equipment. To evaluate the thermal grinding performance, the fraction of coarse and concrete fines based on 1mm particle size was measured, and the amount of residual cement in each part was analyzed by wet analysis using 4M hydrochloric acid. The result was compared with previous studies and the thermomechanical equipment could be selected to enhance the process. Therefore, it is expected that the equipment for commercialization could be optimized and composed the process compactly by this study.

Acknowledgements

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Keywords: Radioactive concrete, Volume reduction, Integrated equipment, Thermomechanical process

Considerations on Safety for Identifying and Preventing the Industrial Hazards During Operation of Induction Melting Furnace

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Working with molten metal has always been and will always be a dangerous workplace. No matter how carefully equipment is designed, workers are trained and procedures are followed, the possibility of an accident can occur in melting workplace.

Some primary causes of melt splash and furnace eruptions include wet or damp charge material, dropping heavy charge into a molten bath, wet or damp tools or additives and sealed scrap or centrifugally cast scrap rolls. Induction melting brings together three things (water, molted metal and electricity) that have the potential for concern if the furnace is not properly working. Induction furnace must have a water cooling system built into the coil itself. Water picks up the heat caused by the current as well as heat conducted from the metal through the refractory. The water carries the heat to a heat exchange for removal. Spill pits serve to contain any molten metal spilled as a result of accident, run out or dumping of the furnace in an emergency. If a leak is suspected at any time, cease operation and clear the melt deck area of all personnel and empty the furnace. Molten metal fins can penetrate worn or damaged refractory and come into contact with the coil. A furnace or a close capture hood which suddenly swings down from a tilted position will cause injury or death. Whenever workers are working on a furnace or close capture hood when it is in the tilted position, be sure that it is supported with a structural brace that is strong enough to keep it from dropping if hydraulic pressure is lost. In theory refractory wear should be uniform, however, in practice this never occurs. The most causes of lining failure are improper installation of refractory material, inadequate sintering of refractory material, failure to monitor and record normal lining wear, allowing the lining to become too thin, installation of the wrong refractory, improper preheating of a used cold lining, failure to properly maintain the furnace the sudden or cumulative effects of physical shocks or mechanical stress, and excessive slag or dross buildup. Pouring cradles provide bottom support for crucibles. A crack in the crucible occur below the bottom ring support, the bottom of the crucible can drop and molten metal will spill and splash, possibly causing serious injury or death. To reduce this danger, a pouring cradle that provides bottom support for the crucible must be used. Power supply units must have safety locks and interlocks on all doors and access panels. Workers who work with low voltage devices must be made aware of the risk posed by high levels of voltage and current.

The most causes of accidents are introduction of wet or damp material, improper attention to charging, failure to stand behind safety lines, coming into contact with electrically charged components and lack of operator skills and training. Only trained and qualified personnel are to have access to high risk areas. Safety lockout systems are another effective measure to prevent electrical shock.

Keywords: Accident, Induction furnace, Industrial hazard, Melting, Safety

Considerations on Event Sequence and Consequences of Critical Hazards Generated From Operation of a Melting Facility for Metal Radioactive Waste

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The critical hazards generated from operation of a melting facility for metal radioactive waste are mainly assumed to be such as vapor explosion, ladle breakthrough and failure in the hot-cell or furnace chamber using remote equipment.

In case of vapor explosion, material containing moisture and/or enclosed spaces may, due to rapid expansion of gases when heated, cause an explosion and/or violent boiling. The rapid expansion of gases may lead to ejection of molten radioactive metal from the furnace into the furnace hall. If there is a large amount of liquid the explosion may damage or destroy technical barriers such as facility walls. The consequences for the facility ranges from relatively mild to very severe depending on the force of the explosion as well as the type of waste being melted. Non-radiological consequences may be physical damage or destruction of equipment and facility barriers, such as walls. Due to the radiological consequences a longer operational shutdown would likely be required. Cleanup efforts would include cutting of solidified metal in a problematic radiological environment requiring use of remote technology before damage and repair requirements can be assessed. Even though there is a risk for direct physical harm to operators for example in the control room and hot-cell, this analysis focuses mainly on the radiological impact. The extent to which remote equipment could be used in the decontamination effort will largely determine the health consequences to the workers. It is reasonable to assume that there will be a need for workers to participate manually in the effort. Due to the potentially large dose rates and the physical environment, it is possible that the maximum allowable dose burden to a worker will be reached. No major consequence for the environment is expected as most of the radioactivity is bound to the material.

In case of ladle breakthrough, a ladle breakthrough involves loss of containment of the melt due to damage of the ladle. This may be caused e.g. by increased wear due to overheating in the melt, or from physical factors such as mechanical stress and impact from the waste. A ladle breakthrough may lead to spread of molten metal in the furnace hall. Molten metal coming into contact with the surrounding cooling equipment may cause a steam explosion. The consequences of a ladle breakthrough will depend on the event sequence. The most severe is when the molten metal comes into contact with the cooling system causing a vapor explosion. The basic consequences are assumed to be similar to those of the vapor explosion above. While the ejection of molten metal is likely more local in the ladle breakthrough scenario, the consequences are judged to be similar.

In case of failure in the hot-cell or furnace chamber using remote equipment, the loss of electric supply or technical failure in the furnace causes loss of power supply. If not remedied quickly, this could lead to that the melt solidifies. A melt that is solidified due to cooling after loss of power cannot be removed nor re-melted. This may occur especially fast if there is not melted material in the furnace. An unscheduled replacement of the refractory in the furnace would be required. It could be unknown to what degree remote equipment can be used to cut a solidified melt. It is therefore assumed that personnel may need to be employed. This event could not have any impact on environment.

Keywords: Hazard, Melting, Metal, Radioactive waste

Characteristic Evaluation of GAGG Detector in High-Radioactivity Area

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Radioactive contamination distribution in nuclear facilities is typically measured and analyzed using radiation sensors. Since generally used detection sensors have relatively high efficiency, it is difficult to apply them to a high radiation field. Therefore, shielding/collimators and small size detectors are typically used. Nevertheless, problems of pulse accumulation and dead time still remain. This can cause measurement errors and distort the energy spectrum.

In this study, this problem was confirmed through experiments, and signal pile-up and dead time correction studies were performed. A detection system combining a GAGG sensor and SiPM with a size of $10\text{ mm} \times 10\text{ mm} \times 10\text{ mm}$ was used, and GAGG radiation characteristics were evaluated for each radiation dose ($0.001\sim 57\text{ mSv/h}$). As a result, efficiency increased as the dose increased, but the energy spectrum tended to shift to the left. At a radiation dose intensity of 400 Ci (14.8 TBq), a collimator was additionally installed, but efficiency decreased and the spectrum was distorted. It was analyzed that signal loss occurred when more than 1 million particles were incident on the detector. In this high-radioactivity area, quantitative analysis is likely to be difficult due to spectral distortion, and this needs to be supplemented through a correction algorithm. In recent research cases, the development of correction algorithms using MCNP and AI is being actively carried out around the world, and more than 98% of the signals have been corrected and the spectrum has been restored. Nevertheless, the artificial intelligence (AI) results were based on only 2-3 overlapping pulse data and did not consider the effect of noise, so they did not solve realistic problems. Additional research is needed.

In the future, we plan to conduct signal correction research using $\approx 10 \times 10\text{ mm}$ small size detectors (GAGG, CZT etc.). Also, the performance evaluation of the measurement/analysis system is intended to be performed in an environment similar to the high radiation field of an actual nuclear facility.

Keywords: High-radioactivity area, GAGG, Pile-up, Dead time, Signal correction

Study on the Blasting Decontamination for Very-low-level Metal Waste

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As unit 1 of Kori was permanently shut down in June 2017, domestic nuclear industry has entered the path of decommissioning. The most important thing in decommissioning is cost reduction. And volume reduction of radioactive waste is especially important. According to the IAEA report, more than 4,000 tons of metallic waste is generated during the decommissioning of a 1,000 MWe reactor and most of these wastes are LLW or VLLW. To reduce amount of metallic waste dramatically, we should choose efficient decontamination method. In this study, we conducted dry ice and bead blasting decontamination.

We prepared Inconel-600 and STS-304 specimen with dimensions of 30 mm × 30 mm × 5 mm. Loose and fixed contamination was applied on the surface of specimen using SIMCON method. Bead and dry-ice blasting was conducted by spraying alumina and dry ice pellet at the same pressure and distance for the same time.

The removal of loose contamination was observed using microscope. It was found that contaminants are significantly removed using both dry ice blasting and bead blasting. However, some abrasive material remained on the surface of specimen. The removal of fixed contamination was verified by weight comparison before and after experiment and cobalt concentration comparison before and after experiment using X-ray Fluorescence Spectroscopy (XRF). At least 90% of the cobalt was removed, but some abrasive particle was also remained on the surface of specimen.

In this study, it is confirmed that the effectiveness of manufacturing a large-scale abrasive decontamination facility, and it is expected that this technology can be used to effectively reduce the amount of metallic waste generated during decommissioning.

Acknowledgements

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Keywords: Blasting decontamination, Radioactive waste, Suitability for disposal, Grit-bead, Dry-ice

Preparation and Characterization of Acrylic Fiber-Based Chelate Resin Modified With Diethylenetriamine

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Chelate resin is a resin that has an exchange group which can form chelates with various metal ions. It shows higher selectivity for metal ions than ion exchange resin and can selectively remove characteristic metal ions. In an aqueous solution containing metal ions, chelate resin can adsorb specific metal ions, and the separated chelate resin can desorb the adsorbed metal ions by changing temperature or pH, so chelate resin has the advantage of being reusable. Chelate resin has been used industrially as an adsorbent to adsorb and separate heavy metal ions in wastewater, and is also used for the purpose of recovering precious or rare metals contained in industrial wastewater or industrial waste. Against this background, there is a need to develop chelate resins with higher adsorption capacity.

Acrylic fiber is defined as a man-made fiber made from a linear synthetic polymer with fiber-forming ability consisting of more than 85% acrylonitrile. It is a man-made fiber that is often used as a substitute for wool because it has good thermal insulation properties like wool and is warm and soft to the touch. It is a fiber rich in cyano groups due to its high content of acrylonitrile, and has the advantage of being able to be used as a variety of functional fibers through modification of cyano groups.

In this study, the amination reaction of acrylic fiber was performed using diethylenetriamine, and the adsorption characteristics for metal ions were evaluated according to the reaction conversion rate. In order to improve the amination efficiency, 400 kGy was irradiated using a 2.5 MeV electron beam accelerator, and through this, the crosslinking rate of acrylic fiber was able to be improved up to 80%. Water and ethanol were used as cosolvents for the amination reaction in a ratio of 60/40 vol/vol, respectively, and a reaction yield of 178% was obtained after 120 minutes of reaction. Using the chelate resin prepared in this way, the adsorption performance for metal ions was evaluated through Atomic Absorption Spectrometry analysis.

Keywords: Chelate resin, Acrylic fiber, Diethylenetriamine, Metal adsorption

Partially Hydrolyzed Poly (Vinyl Acetate)-Borax Complex for the Removal of Simulated Nuclear Fallout Particles

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A Partially hydrolyzed poly (vinyl acetate) (PHPVA)-borax complex-based gel-like coating was successfully developed for the decontamination of Simulated nuclear fallout (SFO) from surfaces. The sprayable coating was self-generated on the surface by borate-diol ester bonds after simultaneously mixing two solutions of borax and PHPVA. The SFO particles, synthesized at 1,200°C for melting, were glassy while some crystalline phases (e.g., SiO₂ and Fe₂O₃) existed together. The SFO particles were fixed onto the Stainless steel (SS) substrate by dropping and evaporating water. for examination of the dust-removal performance of PHPVA-borax based coating. The dusts on the SS surface was successfully removed by casting the PHPVA-borax based coating within 1 minute, demonstrating the excellent dust-removal property of the PHPVA-borax based coating. The used PAB complex in wet state was recovered by using vacuum suction machine in short time. The solid-state PHPVA-borax based film was self-delaminated from the SS substrate after fully drying the used PHPVA-borax coating but this requires long period of time.

Acknowledgements

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Keywords: Partially hydrolyzed poly (vinyl acetate)-borax complex, simulated nuclear fallout particles, Surface decontamination, Gel-like coating

Removal of Radioactive Cesium From Surfaces Using Spray Coating Based on the Phenylboronic Acid-diol Ester Bonds

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Wide-area surface decontamination is essential during the sudden release of radioisotopes to the public, such as nuclear accidents or terrorist attacks. A spray coating composed of a reversible complex between poly (vinyl alcohol) (PVA) and phenylboronic acid-grafted poly (methyl vinyl ether-alt-mono-sodium maleate) (PBA-g-PMVE-SM) was developed to remove radioactive cesium from surfaces. The simultaneous spray of PVA and PBA-g-PMVE-SM aqueous polymer solutions containing Cs adsorbent to contaminated surfaces resulted in the spontaneous formation of a PBA-diol ester bond-based gel-like coating. The Cs adsorbent suspended in the gel-like coating selectively removed Cs-137 from the Cs-contaminated surface. The used gel-like coating were removed from surfaces by simple water rinsing. This recovery way has advantages compared with costly incineration to remove the organic materials for final disposal/storage of the radioactive waste. Thus, our spray coating is suitable for practical wide-area surface decontamination. In radioactive tests, the hydrogel containing Cs-adsorbent showed substantial Cs-137 removal efficiencies of 96.996% for painted cement and 63.404% for cement, which are 2.33 times better than the values for the commercial surface decontamination coating agent DeconGel.

Acknowledgements

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Keywords: Surface decontamination, Cs removal, Spray coating, Phenylboronic acid-diol ester bonds

A General Approach to Planning and Implementing the Environmental Remediation of Radioactively Contaminated Sites

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General phases in the plan and implementation of an environmental remediation of radioactively contaminated sites are planning for remediation, site characterization, remediation criteria, remediation strategy, implementing remediation actions, and conducting post-remediation activities. Environmental remediation should commence with a planning stage. It is helpful to prepare reports which detail all the supporting activities related to these elements before significant levels of funds and efforts are committed. Site characterization is needed to provide sufficient data to make strategic decisions on the environmental remediation activities. The source characterization should include both waste characterization and facility or site characterization and should provide reliable estimates of the release rates of radioactive constituents as well as constituent distribution. During the preliminary site characterization, an engineering study should be conducted to develop remediation options which address the specific contaminant problem and are aimed to reduce radiological and chemical exposure. Options will include engineering approaches and associated technologies. A preliminary selection of options may be made based on several factors including future sites use, technical considerations, public acceptability, cost, and regulatory requirements. The implementation of remediation actions includes procurement of the selected technology, preparation of the site, development of a health and safety plan, development of operations procedures, staff selection and training, completion of site cleanup, verification, waste disposal, and release of the site for any future use. Once remediation activities have been completed and verified, the remediated site can be released for restricted or unrestricted use. Remediation of radioactively contaminated sites may require special adaptation to address sites covering very large surface areas or those which are deep and difficult to access. Quality assurance may be very important to the verification of environmental remediation activities.

The selection of optimal remediation technologies to solve or mitigate the safety of an environmental contamination problem should be taken into account several factors. The several factors include performance (the ability of the technology to reduce risk to the health and safety of the public and to the environment), reliability and maintenance requirements for the technology, costs of implementing the technology, infrastructure available to support the technology, availability (the ease of accessing the technology and associated services), risk to workers and public safety, environmental impacts of the technology, ability of the technology to meet regulatory acceptance, and communication of stakeholder.

Keywords: Remediation, Radioactively contaminated sites

A Review on Types of Induction Melter in Radioactive Treatment Waste Facilities

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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. Also, in the case of Metal radioactive waste, it is necessary to calculate the generation amount in order to treat radioactive waste at a Radioactive Waste Treatment Facility (RWTF).

Since a large quantity of metal radioactive waste is generated during the decommissioning of a nuclear power plant, the application of a metal melter for reduction is considered. The metal waste is heated to a temperature above the melting point and separated into liquid and gas forms. Nuclides existing on the surface of metal waste vaporize in a melting furnace to become dust or collect in sludge. Nonvolatile nuclides such as Co, Fe and Mn remain in ingot, but other nuclides can be captured and reduced with dust and sludge. And the types of melting furnaces to be applied can be broadly classified into Atmospheric Induction Melter (AIM) and Vacuum Induction Melter (VIM).

Therefore, this review intends to compare the two types of metal furnaces to be included in RWTF.

Keywords: Decommissioning, Metal radioactive waste, Decontamination, Radioactive waste

In-Depth Review of Factors Affecting Gamma-Scintillation

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There are analytical methods used for measuring activity when light photons are emitted for scintillating-based analytical application. When this electron returns to the original stable state, it releases its energy in the form of light emission (visible light or ultraviolet light), and this phenomenon is called scintillation. Scintillator is a general term for substances that emit fluorescence when exposed to radiation such as gamma-rays. Radioactivity is all around us and is unavoidable because of the ubiquitous existence of background radiations emitted by different sources. The scintillator contributes to these sensing, and it is expected that the inspection accuracy and limit of detection will be improved and new inspection methods will be developed in the future. Moreover, scintillators are chemical or nanomaterial sensors that can be used to detect the presence of chemical species and elements or monitor physical parameters on the nanoscale. In this study, it includes finding use in scintillating-based analytical sensing applications. A chemical and nanomaterial based sensors are self-contained analytical tools that could provide information about the chemical compositions or elements of their environment, that is, a liquid or even gas condition. Herein, we present an insightful review of previously reported research in the development of high-performance gamma scintillators. The major performance-limiting factors of scintillation are summed up here. Moreover, the 2D material has been discussed in the context of these parameters. It will help in designing a prototype nanomaterial based scintillators for radiation detection of gamma-ray.

Acknowledgements

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Keywords: Gamma-ray, Radiation detection, Scintillation, Nanomaterial

Chemical Decontamination Technologies for the Decommissioning of NPPs

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The radiation field generated in the primary cooling system of a nuclear power plant tends to increase in intensity as radionuclides bind to the oxide film on the internal surface of the primary system, which is operated at high temperature and pressure, and as the number of years of operation increases. Therefore, decontamination of the primary cooling system to reduce worker exposure and prevent the spread of contamination during maintenance and decommissioning of nuclear power plants uses the principle of simultaneous elution of radionuclides when the corrosion oxide film dissolves. In general, a multi-stage chemical decontamination process is applied, taking into account the spinel structure of the corrosion oxide film formed on the surface of the primary cooling system, i.e. an oxidative decontamination step is applied first, followed by a reductive decontamination step, which is repeated several times to reach the desired decontamination goal. Currently, permanganic acid is commonly used in oxidative decontamination processes to remove Cr from corrosion oxide films. In the reductive decontamination step to remove iron and nickel, organic acids such as oxalic acid are commonly used. However, organic acids are not suitable for the final radioactive waste form. A number of multi-stage chemical decontamination technologies for primary cooling systems have been developed and commercialized, including NP-CITROX, AP/NP-CANDECON, CANDERM, AP/NP-LOMI and HP/CORD-UV. Among these, HP/CORD-UV is currently the most actively applied primary cooling system chemical desalination process in the world. In this study, KAERI has developed a new chemical decontamination technology that does not contain organic chemical decontamination agents, with a focus on securing an original technology for reducing the amount of decontamination waste while having equivalent or better decontamination performance than overseas commercial technologies, and compared it with the inorganic chemical agent-based HyBRID (Hydrazine Based Reductive Metal Ion Decontamination) chemical decontamination technology.

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Keywords: Chemical decontamination, Primary cooling system, Oxide layer, HyBRID

Investigation of Corrosion Inhibitors for Chemical Decontamination

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It is crucial to be sure about the safety of nuclear facilities for human resources who are in danger of radioactive emission, also diminishing the volume of the wastes that are buried under the ground. Chemical decontamination of nuclear facilities can provide these demands at the same time by dissolving the oxide layer, which radionuclides such as ⁶⁰Co and ⁵⁸Co have been penetrated, of parts that are utilized in nuclear plants. Although there are many commercial methods to approaching its aim and they perform a high decontamination factor, they have some issues such as applying organic acids which have the ability to chelate with radionuclides that can be washed by underground water, have large quantities of radioactive waste and damage to the surface by severe intergranular attack. A new method has been introduced by KAERI's scientist which is named the HyBRID Process, in this process the main solution is the acidic form of Hydrazine. In this process, like other acid-washing processes, there is a chance of corrosion on the metal surface which is not desired. The metal surface is able to be protected during dissolving process by adding some organic and inorganic corrosion inhibitors such as PP2 and PP3. There is a very new research topic about ionic liquids (ILs) as corrosion inhibitors which illustrates a vast potential for this application due to their tunable nature and the variety of options for cationic and anionic parts. The key factors for ILs corrosion inhibitors such as the hardness properties are summarized. In this study, we review to the fundamentals and development of corrosion inhibitors for chemical decontamination and give an prospect with emphasis on the challenges to be overcome.

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Keywords: Chemical decontamination, HyBRID process, Corrosion inhibitor, Ionic liquids

A Study on Methods to Prevent Contamination From Spreading When Dismantling Calandria Vessel in PHWR NPPs

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Wolsong Unit 1 nuclear power plant, which was permanently shut down in 2019, has a 678 MWe calandria vessel of the CANDU-6 type pressurized heavy-water reactor model.

The calandria inside the vault is a horizontal cylindrical vessel made of stainless steel with a length of 7.8 m and a thickness of 28.6 mm.

For the entire dismantling processes of a nuclear power plant, dismantling works cannot be performed using only one cutting technology and method, and when performing dismantling of a calandria vessel, various systems and components can be used for cutting and dismantling.

The calandria vessel is located in a concrete compartment called a vault, and in order to safely dismantle the calandria vessel, the spread of radioactive contaminants from inside of the vault to the outside must be prevented.

We designed dismantling processes using the laser cutting method to dismantle the calandria vessel and end shields. We must minimize the risk of internal radiation exposure to workers from aerosols derived from the thermal cutting processes. Therefore, we need a way to prevent secondary contamination from spreading outside the vault and within the reactor building.

The path through which radioactive contaminants move is that the flying airborne products generated during the cutting process inside the vault where the calandria is located do not stay in place but spread outward through the opening of the RM-Deck structure at the top.

Therefore, facilities or devices are needed to effectively prevent the spread of radioactive contaminants by blocking the expected movement path.

By using these facilities or devices, it is possible to prevent the movement of radioactive aerosol particles between the location of the worker and the location of the cutting area where the calandria is located, thereby preventing internal exposure through the worker's breathing. In addition, by using these, the cutting area where airborne pollutants are generated can be designed as an isolated work space to prevent the spread of radioactive contaminants.

In this study, we propose a method of facilities for confining radioactive aerosol particles and preventing the spread of contamination when thermal cutting of the calandria vessel within the vault.

Keywords: Decommissioning, Radioactive waste, Dismantling, Contamination, Radiation protection

Preparation and Characterization of Hydrophobic Pt Catalysts Using H₂O₂ Treated Black Carbon for Hydrogen Isotope Separation

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Pt/C catalysts were prepared using black carbon (CB), and evaluated for their potential application as a catalyst of liquid-phase catalytic exchange for tritium treatment. CB was treated with 10% H₂O₂ solution for 0 and 2 hours at 105°C, Ethylene glycol and 40wt% Pt were added to the dried treated sample to prepare a Pt/C catalyst. The physical and chemical properties of the prepared catalysts were evaluated by BET, XRD, elemental analysis (EA), and TEM analyses. As a result of BET analysis, the surface area of CB without 10% H₂O₂ was 237.2 m²·g⁻¹, and after treatment with 10% H₂O₂, it decreased to 181.2 m²·g⁻¹ for 2 hours. However, the internal surface area increased, indicating the possibility that more Pt could be distributed inside the CB treated with 10% H₂O₂. In the XRD analysis results, the presence of Pt was confirmed by observing the Pt peak in the prepared Pt/C catalyst, and it was also observed through TEM analysis that Pt was evenly distributed within the CB. The elemental analysis (EA) results showed that the ratio of S and N decreased and the ratio of O increased with increasing 10% H₂O₂ treatment time. The H₂O₂ treated carbon supported Pt catalysts and polytetrafluoroethylene were then loaded together on a foamed nickel carrier to obtain hydrophobic catalysts. Our hydrophobic Pt catalyst using H₂O₂ treated black carbon are expected to be usefully used in the tritium treatment system.

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Keywords: Black carbon, Pt/C, Catalysts, Liquid-phase catalytic exchange, Tritium treatment

Analysis of the Final Status Survey (FSS) Procedure for Zion Nuclear Power Plant

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The purpose of this report is to provide a summary of the Phase 1 Final Status Survey (FSS) Final Report results and overall conclusions which conduct that the Zion Nuclear Power Station (ZNPS) facility and site meets the 25 mrem(0.25 mSv)per year release criterion as established in Nuclear Regulatory Commission Regulation (NRC) 10 CFR 20.1402 “Radiological Criteria for Unrestricted Use”. The FSS results provided assessment and summarize that any residual radioactivity results in a Total Effective Dose Equivalent (TEDE) to an Average Member of the Critical Group (AMCG) that does not exceed 25 mrem per year, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). The release criterion is translated into site-specific Derived Concentration Guideline Levels (DCGLs) for assessment and summary. ZionSolutions, a decommissioning service provider, estimates that a total of four (4) FSS Final Reports be generated and submitted to the NRC during the decommissioning project.

ZionSolutions established the Characterization/License Termination (C/LT) Group, within the Radiation Protection division, with sufficient management and technical resources to fulfill project objectives. The C/LT Group is responsible for the safe completion of all surveys related to characterization and final site closure. Approved site procedures and detailed Technical Support Documents (TSD) direct the FSS process to ensure consistent implementation and adherence to applicable requirements. The development and planning phase was initiated in 1999 by the “Zion Station Historical Site Assessment” (HSA) and the initiation of the characterization process for FSS.

Develop the information necessary to support FSS design, including the development of Data Quality Objectives (DQOs) and survey instrument performance standards. DQOs are qualitative and quantitative statements derived from the DQOs process that clarify technical and quality objectives.

The next step, FSS design utilizes the combination of traditional scanning surveys, systematic sampling protocols and investigative/judgmental methodologies to evaluate survey units relative to the applicable release criteria for open land sample plans. To aid in the development of an initial suite of potential radionuclides of concern for the decommissioning of ZNPS, the analytical results of representative characterization samples collected at the site were reviewed. At this FSS design step, the Radionuclides of Concern (ROC) are determined. As Co-60 and Cs-137 account for 99.5% of the analysis results of concrete core sampling data from ZNPS’s Containment Building and Auxiliary Building, they are determined and used as the basic ROC in the survey design. Additionally, site information is described and Historical Site Assessment (HSA) is performed. Data collected for the initial HSA will be used to establish the initial regional survey unit and corresponding MARSSIM classification. Next, an assessment of the collected data is performed using the DQO process, and a survey methodology is established by selecting a sampling method and measuring instrumentation. These result judgments provide guidance for C/LT Engineer to interpret findings using the Data Quality Assessment (DQA) process, which analysis Recorded data, Missing values, Deviation from established procedure, and Analysis flags. In conclusion, FSS is the process used to demonstrate that the ZNPS facility and site comply the radiological criteria for unrestricted use specified in 10 CFR.20. The purpose of FSS Sample Plan is to describe the methods to be used in planning, designing, conducting, and evaluating the FSS.

Keywords: Decommissioning, Zion nuclear power station (ZNPS), Unrestricted area

Exposure Dose Assessment and Preliminary Evaluation of Derived Concentration Guideline Level for Potentially Contaminated Surface Soil at Wolsong NPP Site Using RESRAD-OFFSITE Code

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For the release of the nuclear power plant site after the decommissioning, a reliable exposure dose assessment considering the environmental impact of residual radionuclides is essentially required. In this study, the Derived Concentration Guideline Level (DCGL) for the hypothetically contaminated surface soil at the Wolsong nuclear power plant (NPP) unit 1 site was preliminarily calculated by using the RESRAD-OFFSITE computational code and compared with the other case studies. Moreover, radiation exposure dose for local residents and relevant exposure pathways were quantitatively analyzed based on the calculation model established through this work.

For the target site modeling, the source term was determined by referring to the previous case studies regarding the nuclear power plant decommissioning, quantification analysis data of pressure tubes of Wolsong NPP unit 1, and radionuclide data estimated by using the MCNP/ORIGEN-2 code. In total, 14 different radioisotopes such as Ag-108m, C-14, Co-60, Cs-134/137, Fe-55, H-3, Nb-93m/94, Ni-63, Sb-125, Sn-121m, Sr-90, and Zr-93 were considered as target radionuclides. In addition, the geological structure model of the Wolsong NPP site was established based on the final safety analysis report of Wolsong NPP unit 1. The distribution coefficients (K_d) were taken from the JAEA-SDB to estimate the migration/retardation behavior of various radionuclides under the groundwater condition of the Wolsong NPP site. In the present work, the DCGL values were calculated according to the site release criterion of 0.1 mSv/yr, which indicates the radiation protection standard for the site release. Moreover, the exposure pathway and sensitivity analyses were conducted to assess the sensitive input parameters remarkably influencing the calculation result. For the evaluation of exposure dose for local residents, a site layout centered around Wolsong NPP unit 4, located in the closest proximity to the residents' habitation area, was alternatively established and all potential exposure pathways were considered as a comprehensive resident farmer scenario.

The results obtained from this study are expected to serve as a preliminary case study for the DCGL values regarding the surface soil at the Wolsong NPP unit 1 site and for evaluating the radiation exposure dose to local residents resulting from the residual radioactivity at the site after the decommissioning.

Keywords: Wolsong NPP, Decommissioning, RESRAD-OFFSITE, DCGL

A Case Study on Demonstration of Compliance of Dose Criteria for Site and Building Reuse of Foreign Cyclotron Facilities

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For safe and successful decommissioning, it is one of the most important procedures that establishing the goal and complying with regulations of which final status of decommissioned site and building. The dose criteria for cyclotron facilities should be established and applied to reuse the site and building, since building and component of a cyclotron facility have been activated by incident secondary neutrons from radioactive isotope processes (e.g. $^{18}\text{O}(p,n)^{18}\text{F}$, etc.). Furthermore, appropriate approaches should be applied to demonstrate compliance with the dose criteria for reliability of reuse.

It is of noted that U.S. NRC (Nuclear Regulatory Commission) has confirmed that the residual radioactivity which distinguishable from background radiation results in a TEDE (Total Effective Dose Equivalent) does not exceed 25 mrem (0.25 mSv) per year as radiological criteria for unrestricted use of not only nuclear power plants but also cyclotron facilities referred to 10 CFR Part 20.1402. In addition, U.S. NRC noted the two approaches (i.e. dose assessment methods and, DCGL and final status surveys) which can be applied for demonstrating compliance with the dose criteria of 10 CFR Part 20 and recommended DCGL and FSS approach based on advantages and disadvantages of the two approaches. In order to using DCGL and FSS approach, U.S. NRC suggested screening approach; using DandD Version 2 which assesses TEDE under ICRP 28 and site-specific approach; using all models or computational codes which approved by NRC staff.

There are several foreign cases that release of cyclotron facilities after decommissioning (i.e. U.S. and Japan). U.S., for examples, there are two DCGL approach cases and one dose modeling case based on 25 mrem per year same as reactor facilities. The dose modeling case, however, which may not be really used in Korea because of its low applicability. On the other hand, Japan case did not establish any radiological criteria for site and building reuse such as DCGL and just confirm “no more contamination” which is all residual radioactivity is lower than MDC based on real survey. Japan case also may not be used in Korea since criteria of “no more contamination” is not clear and hard to apply for all sites.

Considering regulations and criteria for site release and reuse in Korea, this study aims to suggest radiological criteria and the demonstration approach of compliance for decommissioning of cyclotron facilities based on Nuclear Safety Acts and NSSC notices.

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Keywords: Decommissioning, Site release, Site reuse, Dose criteria, DCGL (derived concentrations of guideline levels), Cyclotron

Disassembly of Highly Radioactive Contaminated Equipment in Hot Cell

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Korea Atomic Energy Research Institute (KAERI) has been operating the Post Irradiation Examination Facility (PIEF) for fuel examinations. The facility has pools and hot cells for handling and examining fuel assemblies and rods. Among the hot cells, the second cell is for measuring rod internal pressure (RIP) and then cutting the rod to make samples for destructive tests. Currently, the cutting machine is broken, so it has to be replaced. Because the existing cutting machine consists of many parts and its size was quite a bit large to handle and treat for the radioactive waste disposal, the disassembly work has been performed to make it smaller using manipulators. The drawings of the cutting machine were reviewed and the disassembly tools were developed considering workability when the work performed at the hot cell using the manipulators. The large parts such as motor, mirror and cable, etc., were able to be disassembled and the machine size became so smaller that it could be easily handled for the disposal.

Keywords: Hot cell, Fuel rod, Cutting machine, Post irradiation examination, Fuel cladding

Cost Benefit Analysis of Treatment Process for Spent Filter

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The treatment process for Spent Filter(SF) of Kori-1 was developed that includes the following : 1) Taking out by robot system 2) Screening by ISOCS 3) Collection of representative samples using a sampling machine 4) Compression 5) Immobilization 6) Packaging and nuclide analysis and 7) Delivery/disposal. Although the robot system, ISOCS, sampling machine and immobilization facility are essentially required for building the above processing but decision to build the compression system and nuclide analysis system must be made after reviewing the need and cost benefit for their construction. In addition, for efficient SF treatment, it is necessary to determine the nuclide concentration range of the SF to which immobilization will be applied. In this study, a cost benefit analysis was performed on existing and alternative methods for processes related to compression treatment, nuclide analysis and immobilization methods, which are greatly affected by economics and efficiency according to the design.

First, although the disposal cost is reduced with reducing the number of packaging drums by compressed and packaged but the expected benefits not be equal to or greater than the cost invested in building a compression system. As a result, non-compressed treatment of SF is expected to be economical because the construction cost of compression system is more expensive than the benefits of reducing disposal costs by compression. Second, a cost benefit analysis of direct and indirect nuclide analysis methods was performed. For indirect analysis, scaling factors should be developed and the drum scanner suitable for the analysis for DAW should be improved. As a result, direct analysis applied grouping options is expected to be more economical than indirect analysis requiring the cost for developing scaling factors and improving the scanner. Third, it is time-consuming and inefficient to distinguish and collect filters that are subject to be immobilized according to the waste acceptance criteria among the disorderly stored SFs in the filter rooms. If the benefits of immobilization of the SFs selectively are not greater than the benefits of immobilization of all SFs, it can be economical to immobilize all SFs regardless of the nuclide concentration of them. As a result, it is more economical to immobilize all SFs with various nuclide concentrations than to selectively immobilize them.

The conclusion of this study is that it is not only cost-effective but also disposal-effective to design the treatment process of SF to adopt non-compressed processing, direct analysis and immobilization of all SFs.

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Keywords: Spent filter, Cost benefit analysis, Compression, Nuclide analysis, Immobilization

Evaluation of Absorption Dose for PE-HIC

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According to acceptance of radioactive waste, homogeneous waste such as concentrated liquid waste and spent resin must be solidified to reduce radiological hazards and protect public health and the ecology. However, when using a High Integrity Containers (HIC), it is stated that homogeneous waste can be disposed of without applying the solidification test requirements. PC-HIC, developed in Korea, is composed of polyethylene (PE, interior), polymer concrete (PC, filler), and steel (external reinforcement). Currently, PC-HIC will be used as a packaging container for low-level liquid waste and spent resin waste. PE has a lower shielding efficiency compared to PC, but it offers the economic advantage of lower production costs. Therefore, cost savings can be expected if very low-level waste is packaged and disposed of HIC made only of PE materials (PE-HIC).

Despite the economical advantage of PE-HIC, PE-HIC has not been used domestically since NRC (Nuclear Regulatory Commission) reported that PE-HIC couldn't meet the mechanical integrity criteria for radiation exposure. However, according to IAEA (International Atomic Energy Agency) research, it has been reported that mechanical integrity of PE-HIC is not affected when the absorbed dose is below 50 kGy. Therefore, in this study, Radiological impact of VLLW packaged into PE-HIC is evaluated to confirm that the absorbed dose is below 50 kGy, which then be used to assess feasibility of PE-HIC to be used as packaging and disposal container for radioactive waste.

Radiological impact of VLLW packaged into PE-HIC is evaluated to confirm that the absorbed dose is below 50 kGy, which then be used to assess feasibility of PE-HIC to be used as packaging and disposal container for radioactive waste. The feasibility of using PE-HIC as packaging-disposal containers for radioactive waste will be reviewed.

In this study, the radiation effects of only waste packaged in PE-HIC were considered, and additional assumptions are as follows.

- Nuclides subject to radioactivity evaluation: Co-60, Cs-137
- Radioactivity concentration: very low-level radioactive waste concentration limit
- Target waste: waste resin
- PE-HIC dimensions: outer diameter: 1,194 mm, height: 1,290 mm, and thickness 88 mm (PC-HIC internal PE shape)

Considering the above assumption, the exposure rate was evaluated using the MicroShield program. Since the density of PE-HIC in the MicroShield program was assumed as the density of air. The absorbed dose was recalculated through density correction of the derived exposure rate. As a result, it was confirmed that absorbed dose was about 2-3 mGy over 300 years.

As a result of dose evaluation by MicroShield, it is judged that the mechanical integrity of PE-HIC as an packaging of VLLW can be proved by confirming that the absorption dose irradiated to PE-HIC by internal waste is less than 50 kGy.

Acknowledgements

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Keywords: PE-HIC, Absorbed dose, Exposure rate, Radiation, Microshield

The Effects of Thickness and Area of the Contaminated Zone in DCGL Derivation for the NPP Decommissioning

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The Derived Concentration Guideline Level (DCGL) is required to release the facility from the nuclear safety act at the stage of site restoration of the decommissioning nuclear power plant. In order to evaluate DCGL, there are various requirements, and among them, the selection of input parameters based on the application scenario is the main task. Especially, it is important to select input parameters that reflect site characteristics, and at this time, a single deterministic value or a probabilistic distribution can be applied. If it is inappropriate to apply a particular single value, it may be reasonable to apply various distributions, and the RESRAD code provides for evaluation using probabilistic methods. Therefore, this study aims to analyze the difference between the application of the deterministic method and the application of the probabilistic method to the area and thickness of the contaminated zone among the site characteristics data.

This study analyzed the thickness and area of the contaminated zone, and in the case of thickness, the deterministic method was applied by changing the thickness at regular intervals from the minimum depth considered by MARSSIM to the thickness of the unsaturated zone identified in previous research data. In addition, a probabilistic analysis was performed by applying a distribution to the thickness of contaminated zone. Second, for the area of the contaminated zone, the dose was evaluated for each area in consideration of the areas to be considered when deriving Area Factor (AF), and the resulting change in DCGL was observed.

As a result, the DCGL tends to decrease as the thickness increases, and it seems to be saturated when the thickness exceeds a certain thickness. Therefore, It was confirmed that the level of saturated values is similar to that of entering a probabilistic distribution, and in the case of a parameter that is reasonable to enter as a distribution rather than as a single value, it is sufficiently conservative to perform a probabilistic evaluation. In the case of area change, the DCGL evaluation result showed that the DCGL increased as the scale decreased. The magnitude of the change varies depending on the characteristics of each radionuclide, and in the case of radionuclides where external exposure gamma rays have a major exposure effect, the change is relatively small. It can be seen that the change in DCGL according to the area has the same tendency as the AF applicable to the survey unit for small survey units applied in the final status survey.

Keywords: Decommissioning, RESRAD, DCGL, MARSSIM

Analysis of Important Considerations in the Phase of the Demolition Plan of Large Structures When Decommissioning of Nuclear Facilities

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Kori unit 1 and Wolsong unit 1 were permanently shut down in 2017 and 2019, respectively. Both plants were decided to demolish the building without reuse. Large structures must be demolished after removing systems and components in the building, and in the case of large structures, thorough planning is required because of the large scale of work. Therefore, in this study, important considerations in the phase of the demolition plan of large structures when decommissioning were analyzed. The demolition of large structures at nuclear facilities is major one phase of work within a broader decommissioning plan. Furthermore, the actual demolition of the structure (i.e., physical process) represents the last step in a process that begins with extensive planning and analysis. The National Demolition Association (NDA) has provided checklist items that should be considered before the start of a commercial demolition project and/or in the bid process. Important Considerations in the Phase of the demolition plan of large structures when decommissioning of nuclear facilities are Site knowledge and programs, Engineering survey/demolition plan, Hazardous and radioactive materials, Open air demolition, Financial and project management, Permits, Code adherence, and Special programs, Disposal pathway, Final site condition. The results of this study can be used as a basis for the Planning large structures demolition of the Kori unit 1 and Wolsong unit 1.

Keywords: Large structures, Demolition, Planning, Kori unit 1, Wolsong unit 1

A Study on the Direction of Entering the PHWR Decommissioning Market

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The Wolsong unit 1 decommissioning project is the world's first commercial pressurized heavy water reactor decommissioning project. Although there is a lot of accumulated experience and technology for decommissioning of pressurized water reactors around the world, it can be said that there is great difficulty as there is lack of prior experience and reference materials for pressurized heavy water reactor.

On the other hand, if the world's first pressurized heavy water reactor project is completed, it is possible to enter the overseas market for pressurized heavy water reactor decommissioning. It is also a good opportunity to do so.

Accordingly, the current status of operation, plans, and construction of infrastructure related to decommissioning of pressurized heavy water reactors in Canada, which can be said to be the home country of pressurized heavy water reactor, were reviewed.

So, this study seeks to present considerations for entering the pressurized heavy water reactors decommissioning market in the future.

Keywords: Decommissioning, PHWR, Decommissioning strategy, Radioactive waste

Computation Study on the Thermodynamic Energies of HyBRID Dissolution Reaction Considering Cu-Hydrazine Complex Formation

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Decontamination is one of the important processes for dismantling nuclear power plants. The purpose of decontamination is to reduce the radiation levels of contaminated nuclear facilities, ensuring the safety of workers involved in decommissioning and minimizing the amount of radioactive waste.

In this study, we investigate the reaction mechanisms and their thermodynamic energies of the HyBRID (Hydrazine-Based Reductive participated metal Ion Decontamination) process for decontamination of the primary coolant system of a nuclear power plant. We computed the thermodynamic properties of HyBRID dissolution mechanisms in which corrosion metal oxides accumulated in the primary coolant systems along with radionuclides are dissolved by HyBRID decontamination agents ($\text{H}_2\text{SO}_4/\text{N}_2\text{H}_4/\text{CuSO}_4$).

The HyBRID reaction mechanism has been studied using a commercial database (HSC Chemistry®), but Cu ions have been used instead of Cu-hydrazine complexes when calculating reactions due to the absence of thermodynamic properties for Cu-hydrazine complexes.

To address this limitation, we supplemented the quantum calculations with Cu-hydrazine complexes using the density functional calculations. It is intended to simulate a more practical reactions by calculating the reactions considering Cu-hydrazine complexes, and to improve understanding of the HyBRID dissolution reactions by qualitatively and quantitatively comparing the reactions without considering the complex formation.

Keywords: Decontamination, HyBRID process, Cu-hydrazine complex, Density functional theory

Sampling Design of SSCs for Characterization at Shut-Down Power Plant

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The radiological characterization of SSCs (Structure, Systems and Components) plays one of the most important role for the decommissioning of KORI Unit-1 during the preparation periods. Generally, a regulatory body and laws relating to the decommissioning focus on the separation and appropriate disposal or storage of radiological waste including ILW (intermediate level waste), LLW (low level waste), VLLW (very low level waste) and CW (clearance waste), aligned with their contamination characteristics. The result of the preliminary radiological characterization of KORI Unit-1 indicated that, apart from neutron activated the RV (reactor vessel), RVI (reactor vessel internals), and BS (biological shielding concrete), the majorities of contamination were sorted to be less than LLW.

Radiological contamination can be evaluated into two methods. Due to the difficulties of directly measuring contamination on the interior surfaces of the pipe, called CRUD, the assessment was implemented by modeling method, that is measuring contamination on the exterior surfaces of the pipes and calculating relative factors such as thickness and size. This indirect method may be affected by the surrounding radiation distribution, and only a few gamma nuclides can be measured. Therefore, it has limitation in terms of providing detailed nuclide information. Especially, α and β nuclides can only be estimated roughly by scaling factors, comparing their relative ratios with the existing gamma results. To overcome the limitation of indirect measurement, a destructive sampling method has been employed to assess the contamination of the systems and component. Samples are physically taken some parts of the systems or components and subsequently analyzed in the laboratory to evaluate detailed nuclides and total contamination. For the characterization of KORI Unit-1, we conducted the radiation measurement on the exterior surfaces of components using portable instruments (Eberline E-600 SPA3, Thermo G20-10, Thermo G10, Thermo FH40TG) at BR (boron recycle system) and SP (containment spray system) in primary system. Based on these results, the ProUCL program was employed to determine the destructive sample collection quantities based on statistical approach.

The total of 5 and 8 destructive sample quantities were decided by program and successfully collected from the BR and SP systems, respectively. Samples were moved to laboratory and analyzed for the detail nuclide characteristics. The outcomes of this study are expected to serve as valuable information for estimating the types and quantities of radiological waste generated by decommissioning of KORI Unit-1.

Keywords: Decommissioning, Characterization, Contamination, KORI unit-1, Destructive sampling

Corrosion Behaviors of Carbon Steel Related to the Decontamination of Primary Heat Transport System for CANDU Reactor

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The primary heat transport system consists mainly of the in-core fuel channels connected to the steam generators by a system of feeder pipes and headers. The feeders and headers are made of carbon steel. Feeders run vertically upwards from the fuel channels across the face of the reactor and horizontally over the refueling machine to the headers. Structural materials of the primary systems of nuclear power plants (NPPs) are exposed to high temperature and pressure conditions, so that the materials employed in these plants have to take into accounts a useful design life of at least 30 years. The corrosion products, mainly iron oxides, are generated from the carbon steel corrosion which is the main constituent of the feeder pipes and headers of this circuit. Typical film thickness on CANDU-PHWR surface is 75 μm or 30mg/cm². Deposits on PHWR tends to be much thicker than PWR due to use of carbon steel and also for the source of corrosion products available on the carbon steel surface. Degradation of carbon steel for the feeder pipes transferring the primary system coolant by flow-assisted corrosion in high temperature has been reported in CANDU reactors including Point Lapreau, Gentilly-2, Darlington and Bruce NPPs. The formation of Fe₃O₄ film on a carbon steel surface reduces the dissolution rate of steel substantially. The protectiveness of the Fe₃O₄ film over the carbon steel is affected by the environmental factors and the operational parameters of the feeder pipes, including the velocity, wall shear stress, solution pH, temperature, concentration of dissolved iron, quality of solution, etc. For effective chemical decontamination of these thick oxides containing radionuclides such as Co-60, it is necessary to understand the corrosion behaviors of feeder pipes and the characteristics of oxide formed on it. In this work, we investigated the growth of oxide films that develop on type SA-107 Gr. B carbon steel in high temperature water and steam environment by scanning electron microscopy (SEM) and glow discharge optical emission spectrometry (GD-OES) for the quantification and the solid-state speciation of metal oxide films. This study was especially focused to set the experimental tests conditions how to increase the oxide thickness up to 50 μm by changing the oxidation conditions, such as solution chemistry and thermo-hydraulic conditions both temperature and pressure and so on.

Keywords: Nuclear decommissioning, Corrosion behavior, Decontamination, Carbon steel, Magnetite

Speciation of M(II)-Oxalate Complex in an Electrodynamic Cell Dissolving CRUD

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During the operation of a nuclear power plant (NPP), corrosion products called CRUD (Chalk River unidentified deposit) accumulate on the surface of the primary system. The CRUD components of pressurized light water reactors or heavy water reactors, represented by $(\text{Ni}_x\text{Fe}_{1-x})(\text{Fe}_y\text{Cr}_{1-y})_2\text{O}_4$, are composed of Fe_3O_4 , NiFe_2O_4 , FeCr_2O_4 , NiCr_2O_4 , etc. Radionuclide such as Co-60 are deposited within this CRUD, so the entire deposited material must be dissolved and removed for decontamination. Chemical decontamination has the advantage of being able to decontaminate a wide metal surface, but has the disadvantage of generating a large amount of secondary waste.

Recently, chemical decontamination methods that add an electrodynamic process are being studied to overcome these shortcomings. This technology is a method of dissolving CRUD by applying an electric field in the anodic compartment of a cell separated by CEM. It is a method of accelerating CRUD dissolution by generating a large amount of hydrogen ions in the anodic compartment. Dissolved metal ions pass through the CEM (cation exchange membrane) and move to the cathodic compartment ($\text{pH} > 12$), where they are removed by adsorption or precipitation process. Therefore, the speciation characteristics between decontamination agent (oxalic acid) and metal ions are very important.

In this study, we investigated the speciation characteristics of Fe(II), Ni(II), Co(II) - oxalate, which are important complex species in CRUD dissolution cells. The thermodynamic equilibrium constant for hydrolysis of each ion and of M(II)-oxalate were collected and speciation characteristics were analyzed using the MINEQL 5.0 program. From the speciation characteristics of M(II)-oxalate, effective radionuclide removal methods in an electrodynamic cell were considered.

Acknowledgements

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Keywords: Decontamination, CRUD, Oxalic acid, CEM, Electrodynamic process

Adsorption Behavior of Cs(I) and Sr(II) Onto Prussian Blue Functionalized Bentonite

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After the Fukushima accident in 2011, relevant concerns regarding the contamination of the natural environment rose abruptly. For example, water contaminated by radionuclides such as Cs and Sr may directly flow into the ocean and threaten the marine ecosystem. In this respect, cost-effective and efficient decontamination techniques need to be developed and verified to remediate the contaminated water. Prussian blue (PB) is known as a representative material that can adsorb Cs by ion-trapping and is widely used for medical purposes. However, there is a limitation that PB itself is non-separable and highly mobile in aqueous system, so it needs a fixture, such as bentonite, to be collected after the adsorption. Furthermore, while the performance of PB toward Cs is relatively well known, its behavior toward Sr has rarely been reported. The object of this study is to investigate the sorption characteristics of Cs and Sr onto PB-functionalized bentonite at various conditions.

The adsorbent employed in the present work was prepared by mixing bentonite, FeCl_3 , and $\text{K}_4[\text{Fe}(\text{CN})_6]$ at room temperature for 24 hours in the aqueous solution. The concentrations of FeCl_3 and $\text{K}_4[\text{Fe}(\text{CN})_6]$ were set to a range of 5-200 % compared to the cation exchange capacity of bentonite. After that, the PB-functionalized bentonite was sieved with a mesh size of 63 μm and then reacted with the Cs and Sr solution at various liquid-to-solid (L/S) ratios of 2-10 g/L for up to 500 minutes. Moreover, synthetic seawater containing additional Cs and Sr was reacted with PB-functionalized bentonite to characterize the ion selectivity of PB. After the completion of the adsorption experiment, a part of the adsorbent was separated and desorption of Cs and Sr with 2 M of nitric acid was performed. For the quantification of aqueous Cs and Sr concentrations, ICP-MS was employed after the filtration with a pore size of 0.45 μm .

The result obtained in this study revealed a high sorption affinity of Cs and Sr onto PB-functionalized bentonite. The analysis results also presented that the sorption reactions of Cs and Sr reached their steady state within 10 minutes of reaction time. Furthermore, the ion selectivity toward Cs and Sr was verified through sorption test with synthetic seawater. According to the high sorption affinity and selectivity, the PB-functionalized bentonite synthesized through this study is expected to be widely used for remediating the Cs- and Sr-contaminated groundwater and seawater, particularly in nuclear waste-relevant industries.

Keywords: Prussian blue, Functionalized bentonite, Cesium, Strontium, Adsorption

Membrane Electrode Assemblies for High Performance PEM Electrolyzer for Tritium Enrichment

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The development of separation method of radioactive tritium is imperative for treating tritium-contaminated water originating from nuclear facilities. Polymer electrolyte membrane electrolysis technology represents a promising alternative to conventional alkaline electrolysis for tritium enrichment. Nevertheless, there has been limited research conducted thus far on the composition of membrane electrode assemblies (MEAs) specifically optimized for tritium separation, as well as the methods used for their fabrication.

In this study, we conducted an investigation aimed at optimizing MEAs specifically tailored for tritium separation. Our approach involved the systematic variation of MEA components, including the anode, cathode, porous transport layer, and electrode formation method. The water electrolysis efficiency and the H/D separation factor in deuterated water (1%) were evaluated with respect to both the preparation method and the composition of the MEA. To assess the long-term stability of the MEAs, changes in cell voltage, resistance, and the active electrode area were analyzed using impedance analysis and cyclic voltammetry. Furthermore, we examined H/D separation factor both before and after degradation.

The results showed that MEAs with different anode/cathode configurations and electrode formation methods improved the electrolysis efficiency compared to commercial MEAs. In addition, the degree of change in the resistance value was also different depending on the electrode formation method, indicating that the electrode formation method has a significant impact on the stability of the electrolysis system. Therefore, the study showed that the efficiency and long-term stability of the water electrolyzer can be improved by optimizing the MEA fabrication method.

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Keywords: Hydrogen isotope, Stability, Electrode fabrication, PEM (proton exchange membrane), Water electrolysis

A Preliminary Study for CRUD Removal by Multiple Membrane of Electro-Kinetic Process

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When decommissioning of nuclear facilities happens, large amounts of radioactive wastes are released. Because costs of nuclear decommissioning are enormous, effective and economical decontamination technologies are needed to remove radioactive wastes. During NPP operation, corrosion product called Chalk River Unidentified Deposits (CRUD) is generated. CRUD is an accumulation of substances and corrosion products consisting of dissolved ions or solid particles such as Ni, Fe, and Co on the surface of the NPP fuel rod coating. CRUD is slowly eroded by the circulation of hot pressurized water and later deposits on the fuel rod cladding or external housing, thereby reducing heat production by the nuclear fuel. Decontamination of radiologically contaminated metals must be performed before disposal, and several methods for decontaminating CRUD are being studied in many countries.

Decontamination technology is an alternative to reducing human body covering and reducing radioactive waste disposal costs, and much research and development has been conducted to date. Currently, the importance of decontamination is emerging as the amount of waste stored in radioactive waste storage is close to saturation, and the amount of radioactive waste generated must be minimized through active decontamination. In this study, a preliminary study was conducted on the removal of CRUD by multiple membrane in an electro-kinetic process using an electrochemical-based decontamination method. Preliminary research to develop a technology to electrochemically remove CRUD by using a self-produced electrochemical cell to check the pH change over time of the CRUD cell according to voltage, electrolyte, membrane and pH change.

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Keywords: CRUD, Electrochemical decontamination process, Membrane, Electro-kinetic process

Electrochemical Recovery of Cobalt Ions From Aqueous Solution by Using Copper Hexacyanoferrate Electrode

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Heavy metal wastewater containing cobalt (Co^{2+}) has received more attention as an environment issue, which is released from electroplating processes, battery materials industries, nuclear power plants, etc. Especially, cobalt exposed to high-temperature and high-pressure environment during the operation of a nuclear power plant to form corrosion products and forming a chalk river unidentified deposit (CURD) along with radioactive materials generated in cooling water pipes. Cobalt present in the oxide film is mainly Co-60, which emits radiation and causes increased radiation exposure to workers, and efficient management is essential.

In this study, we demonstrated the performance of copper hexacyanoferrate (CuHCF) electrodes in a capacitive deionization (CDI) system for Co^{2+} ions removal. The structure and chemical status of CuHCF used as an electrode material were characterized, and electrochemical properties were evaluated. This study showed that Co^{2+} ions could be efficiently removed in aqueous solutions using CuHCF electrodes. It has been experimentally shown that the ion removal mechanism is driven by the insertion of Co^{2+} ions within the CuHCF lattice channels. The deionization capacities in 20 and 50 $\text{mg-Co}^{2+} \text{ L}^{-1}$ aqueous solutions were 141.62 and 156.85 mg g^{-1} , respectively, and the corresponding charge efficiencies (Λ) were 0.55 and 0.68, respectively. Thus, we suggest that an electrochemically driven process using CuHCF can usefully remove Co^{2+} ions from wastewater.

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Keywords: Cobalt ion removal, Electrochemical, Capacitive deionization, Prussian blue analogues (PBAs)

Optimal Cutting Method of Pipe Connected to Kori Unit 1 Steam Generator

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Kori Unit 1, pressurized water reactor, is the Korea's first commercial nuclear power plant. It successfully generated electricity for a period of 30 years, commencing from April 19, 1978. Following its approval for continued operation in 2008, Kori Unit 1 continued to operate for an additional 9 years, resulting in a total operational period of 39 years. On June 18, 2017, Kori Unit 1 was permanently shut down. Since then, Korea is actively preparing for the decommissioning of nuclear power plant.

During the decommissioning of a nuclear power plant, the heavy components such as reactor, steam generator, pressurizer, reactor coolant pump located in the containment building should be taken out of the containment building. To take out heavy components from the containment building, pipes connected to heavy component should be cut. There are numerous pipes connected to the heavy component, each with varying dimensions and material. Each pipe has a different level of contamination depending on its use. In this study, optimal cutting method of pipe connected to steam generator, one of the heavy components of nuclear power plant, is proposed during the decommissioning of Kori unit 1.

In case of pipe connected to Kori unit 1 steam generator, material is stainless steel or carbon steel. These pipes have varying inner diameter, ranging from 0.6 cm to 74 cm, and thickness ranging from 0.15 cm to 7.1 cm. These pipes are classified as low and intermediate level waste (LILW) or very low level waste (VLLW). Because characteristics of pipes are different, each pipe optimal cutting methods are proposed differently considering material, dimension, contamination level, cutting cost, cutting time, and the management of secondary waste. As a result, the cutting method for pipe of reactor coolant system is selected to orbital cutting. The cutting method of main steam pipe and main feedwater pipe is selected to oxygen cutting. In case of other small pipes, cutting method is selected to circular saw.

Keywords: Nuclear power plant, Decommissioning, Steam generator, Pipe cutting, Cutting method

Study on the Operation Strategies for RCB Ventilation Abnormalities in Decommissioning Nuclear Power Plant

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In nuclear power plant (NPP) decommissioning, ventilation and purification of the building atmosphere are important to create a working environment, ensure worker safety, and prevent the release of gaseous radioactive materials into the environment. The heating, ventilation, and air conditioning (HVAC) system of each building is maintained, modified, or newly installed. In this study, based on APR1400, operation strategies were presented in case of ventilation abnormalities in the reactor containment building (RCB), where highly radioactive particles and high dust are most frequently generated during NPP decommissioning. For research, it was assumed that the entire RCB atmospheric ventilation during decommissioning would use the RCB purge system of the existing NPP and perform continuous ventilation. Additionally, it is assumed that areas where high radiation particles and high dust occur locally, such as reactor containers or internal segments, are sealed with tents and purified using a HEFA filter of a temporary portable HVAC, and a exhaust flow path is connected to the discharge duct of the existing RCB purge system. The possibility of abnormal occurrence was largely divided into two cases. First, when large amounts of uncontrolled pollutants are released into the atmosphere inside the RCB, discharge to the environment is stopped manually or automatically by a modified engineered safety function activation signal (ESFAS). Afterwards, the RCB purge system should be operated in recirculation mode to sufficiently purify the RCB atmosphere with a HEPA filter. Second, when the first train of the low volume purge system is not running due to a failure, standby train should be operated. If both low volume purge trains fail, a high volume purge system is used. Intermittent purge operation is preferred due to large capacity during high volume purge operation. In cases where it is not possible to operate all purge systems due to common issues such as power supply, atmospheric sampling is performed to determine whether to proceed with the work inside RCB.

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Keywords: HVAC, Operation, RCB, Ventilation

A Study on the Validity of Selecting Major Nuclides to Derive Source Terms for Radioactive Corrosion Products Originating From the Surface of Nuclear Fuel During Nuclear Power Plant Decommissioning

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After the major radioactivation structures (RPV, Core, SG, etc.) due to neutron irradiation from the nuclear fuel in the reactor are permanently shut down, numerous nuclides that emit alpha-rays, beta-rays, gamma-rays, etc. exist within the radioactive structures. In this study, nuclides were selected to evaluate the source term for worker exposure management (external exposure) at the time of decommissioning. The selection of nuclides was derived by sequentially considering the four steps. In the first stage, the classification of isotopes of major nuclides generated from the radiation of fission products, neutron-radiated products, coolant-induced corrosion products, and other impurities was considered as a step to select evaluation nuclides in major primary system structures. As a second step, in order to select the major radionuclides to be considered at the time of decommissioning, it is necessary to select the nuclides considering their half-life. Considering this, nuclides that were less than 5 years after permanent suspension were excluded. As a third step, since the purpose of reducing worker exposure during decommissioning is significant, nuclides that emit gamma rays when decaying were selected. As a final step, it is a material made by radiation from the fuel rod of the reactor and is often a fission product found in the event of a Severe accident at a nuclear power plant, and is excluded from the nuclide for evaluation at the time of decommissioning is excluded. The final selected Co-60 is a nuclide that emits high-energy gamma rays and was classified as a major nuclide that affects the reduction of radiation exposure to decommissioning workers. In the future, based on the nuclide selection results derived from this study, we plan to study the evaluation of worker radiation exposure from crud to decommissioning workers by deriving evaluation results of crud and radioactive source terms within the reactor core.

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Keywords: Radioactive corrosion products, Nuclide selecting validity, Decommissioning

Decommissioning Technology Ranging From Radioactive CO₂ Capture to Solidification of Carbowastes

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Carbon 14 (¹⁴C) is radioactive isotope of carbon which emits beta ray with long half-life (5730±30 years). Since the ¹⁴C is significantly hazardous for human being, the appropriate process to treat ¹⁴C is necessary. From the nuclear power plant, the ion exchange resin, graphite, and activated carbon are the main source of ¹⁴C. During the effort to reduce the volume of those wastes, the ¹⁴C is inevitably occurred as carbon dioxide (CO₂) form, so called ¹⁴CO₂. Thus, the development of technology to permanently capture and safely dispose ¹⁴CO₂ is required.

In this presentation, we introduce the decommissioning technology ranging from ¹⁴CO₂ capture to solidification. First, the new class of glass adsorbent is developed which can irreversibly capture CO₂ even under mild conditions. This material promotes the dissolution of alkaline earth ions due to the unstable glass structure. Then, the physical and chemical optimization of glass adsorbent enhances the performance of CO₂ capture. Further, room temperature geopolymeric solidification is also performed to safely dispose ¹⁴C without any potential release.

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Keywords: Glass adsorbent, CO₂ capture, Carbon 14, Radioactive waste, Mineral carbonation, Solidification

A Study on Decommissioning Technology From the EHS Perspective for Safe Decommissioning of the Secondary System of Kori Unit 1 Nuclear Power Plant

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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.

Once the FDP is approved, it is expected that dismantling will begin with the secondary system, and dismantling work on the primary system of Kori Unit 1 will begin after the spent nuclear fuel is taken out. It is expected that the space where the secondary system has been dismantled can be used as a temporary storage place, and the entire dismantling schedule is expected to proceed without delay.

The main equipment of the secondary system is large and heavy. The rotating parts is connected to a single axis with a length of about 40 meters, and is complexly installed over three floors, making accessibility very difficult. A large pipe several kilometers long that supplies various fluids to the secondary system is installed hanging from the ceiling using a hanger between the main devices, and the outer diameter of the pipe is wrapped with insulation material to keep warm.

In nuclear secondary system decommissioning, it is very important to check for radiation contamination, establish and implement countermeasures, and predict and manage safety and environmental risks that may occur when cutting and dismantling large heavy objects. So we plan to evaluate the radiation contamination characteristics of the secondary system using ISOCS (In-Situ Object Counting System) to check for possible radioactive contamination. According to the characteristics results, decommissioning plans and methods for safe dismantling by workers were studied. In addition, we conducted research on how to safely dismantle the secondary system in terms of industrial safety, such as asbestos, cutting and handling of heavy materials and so on.

This study proposes a safe decommissioning method for various risks that may occur when dismantling the secondary system of Kori Unit 1 nuclear power plant.

Keywords: Decommissioning, Dismantling, Secondary system, Turbine building, Pipe, Radiation contamination

Design of a PWR Decommissioning Waste Assessment Program Applying the Work Difficulty Factor

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The domestic Pressurized Heavy Water Reactor (PWR) nuclear power plant, Wolsong Unit 1, was permanently shut down on December 24, 2019. However, research on decommissioning has mainly focused on Pressurized Water Reactors (PWRs), with a notable absence of both domestic and international experience in the decommissioning of PHWRs. If proper business management such as radiation safety and waste is not performed, it can lead to increased business risks and costs in decommissioning. Therefore, the assessment of waste volume and cost, which provide fundamental data for the nuclear decommissioning process, is a crucial technical requirement before initiating the actual decommissioning of Wolsong Unit 1. Decommissioning radiation-contaminated structures and facilities presents significant challenges due to high radiation levels, making it difficult for workers to access these areas. Therefore, technology development should precede decommissioning process assessments and safety evaluations, facilitating the derivation of optimal decommissioning procedures and ensuring worker safety while enhancing the efficiency of decommissioning operations.

In this study, we have developed a program to estimate decommissioning waste amounts for PHWRs, building upon prior research on PWR decommissioning projects while accounting for the specific design characteristics of PHWRs. To evaluate the amount of radioactive waste generated during decommissioning, we considered the characteristics of radioactive waste, disposal methods, packaging container specifications, and the criteria for the transfer of radioactive waste to disposal operators. Based on the derived algorithm, we conducted a detailed design and implemented the program. The proposed program is based on 3D modeling of the decommissioning components and the calculation of the Work Difficulty Factor (WDF), which is used to determine the time weighting factors for each task. Program users can select the cutting and packaging conditions for decommissioning components, estimate waste amount based on the chosen decommissioning method, and calculate costs using time weighting factors. It can be applied not only to PHWRs, but also to PWRs and non-nuclear fields, providing a flexible tool for optimizing decommissioning process.

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Keywords: Decommissioning waste amount, Decommissioning cost, Decommissioning process, Work difficulty factor

A Feasibility Study on Application of Surrogate DCGL to Kori Unit 1 Decommissioning

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Derived Concentration Guideline Levels (DCGLs), which represent the residual radioactivity concentration limits, serve as the pivotal criteria for decontamination during decommissioning of nuclear power plants and are essential for license termination. The analysis of radionuclides in various media to check site-specific and radionuclide-specific DCGLs is a resource-intensive and time-consuming processes, and there are some radionuclides that are hard to analyze.

In the decommissioning of the Rancho Seco nuclear power plant in the United States, a conservative approach was adopted. Potentially highly contaminated areas on the site were identified by collecting and analyzing soil samples, and radionuclides exceeding the Minimum Detectable Concentration (MDC) were selected as the potential Radionuclide of Concern (ROC), and surrogate DCGLs for hard-to-detect radionuclides were applied to soil samples. For soil samples in the Rancho Seco nuclear power plant, Cs-137 contributed more than 90% of the total radioactivity. DCGLs of the ROC were obtained using the scaling factors through analysis of Cs-137 for a large amount of soil samples.

In Korea, the scaling factor methodology has not been applied to the decommissioning of commercial nuclear power plants. An initial investigation was undertaken to assess the viability of implementing Surrogate Derived Concentration Guideline Levels (DCGLs) in the dismantling of Kori Unit 1, drawing insights from the U.S. nuclear power plant decommissioning experiences. To do this approach, the concentration ratio of radionuclides of interest to key radionuclide in contaminated soil should be known and consistent. But related information is not available at this time. So Surrogate DCGL for representative C-14, Fe-55, Ni-59, Ni-63, and Sr-90 was obtained using the scaling factors applied to radioactive waste data, specifically Decontaminated Aqueous Waste (DAW) and Spent Resin.

In order to develop a reliable surrogate DCGLs the Kori Unit 1 site, it is important to analyze the radionuclides in the soil for the Kori Unit 1 decommissioning site to obtain consistent concentration ratio of the radionuclides of concern to the key radionuclides. When a suitable DCGL is developed, it can be used for FSS planning and prior decision-making ensuring the safe and effective decommissioning of Kori Unit 1 and similar nuclear power plants.

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Keywords: Decommissioning, Radionuclide ratio, Surrogate DCGL, Scaling factor

Geometric Modeling and Radioactive Modeling of RVI Using MAVRIC Code

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In 2017, the permanent shutdown of Kori Unit 1 was decided, marking the initiation of preparations for the decontamination and decommissioning of Kori Unit 1. The dismantling of radiologically contaminated equipment and concrete structures such as the Reactor Vessel (RV), Reactor Vessel Internals (RVI), and the Bio shield is crucial in the nuclear decommissioning process. These components became radiologically contaminated due to nuclear fission reactions occurring in the reactor during its operational period.

The RVI dismantling at Spain's Jose Cabrera Nuclear Power Plant involved the use of mechanical saws and disk cutters to divide it into approximately 430 pieces, taking 16 months to complete. Germany's Stade Nuclear Power Plant employed mechanical circular saws to segment their RVI into about 170 pieces, which took 30 months to accomplish. Meanwhile, the RVI at Germany's Wurgassen Nuclear Power Plant was subdivided into approximately 1,200 pieces using a combination of mechanical saws and abrasive water jets, requiring 61 months for completion.

Due to the radioactivity in Kori Unit 1's Reactor Vessel (RV) and Reactor Vessel Internals (RVI), remote-controlled systems were developed for cutting within the cavity to reduce radiation exposure. Specialized equipment was developed for underwater cutting operations.

This paper focuses on modeling related to RVI operations using the MAVRIC code. The upper and lower parts of the RVI are classified as low-level radioactive waste, while the sides of the RVI that come into contact with fuel are classified as intermediate-level radioactive waste. Therefore, the modeling presented in this paper only considers the RVI sides since the upper and lower parts have a minimal impact on radiation exposure. Accurate calculations were performed through geometric modeling and radiation dose modeling.

These research findings are anticipated to contribute to enhancing the efficiency and safety of nuclear reactor decommissioning operations.

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Keywords: MAVRIC, RVI, Modeling, Kori1

Simulated Evaluation of Sludge Accumulation Behavior for Laundry & Hot Shower Tank (LHST) in Nuclear Power Plant

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When dismantling a power plant, a large amount of radioactive tanks are generated, and it is estimated that a significant amount of sludge will accumulate inside the tanks during long-term operation. In the process of dismantling a radioactive tanks, it is important to know the composition of the sludge because the sludge present inside must first be removed and then disposed of.

In the case of certain tanks, it can be predicted that corrosion products generated due to system corrosion are the main cause of sludge formation. However, in the case of some tanks, it is not easy to predict the sludge composition because various dispersed particles in addition to corrosion products may be mixed with the wastewater. Even if it is collected and analyzed, the sludge composition can change significantly depending on the operation history, so the analysis results cannot be considered representative of the composition.

In the case of LHST, surfactant components introduced during the washing and shower process, oil components and dispersed particles dissolved by the surfactant accumulate inside the tank, making sludge difficult to remove. In addition, even if it is removed by ultra-high pressure spraying, unexpected problems may occur in the subsequent treatment process due to the surfactant contained therein. Therefore, it is necessary to analyze in more detail the characteristics of sludge accumulated in LHST and prepare countermeasures.

A test procedure was prepared to evaluate the characteristics of sludge accumulating in LHST. According to the test results, the long-term sludge accumulation tendency of the LHST is summarized as follows.

- ① Initially, the sludge settling speed increases slowly until a surface sludge layer is formed.
- ② After the surface sludge layer is formed, the sludge rapidly settles until the sludge layer becomes somewhat thicker.
- ③ When the sludge layer is formed to a certain extent, the sludge escape rate increases and the sludge accumulation rate decreases again. It is assumed that the sludge escape speed is closely related to the fluid flow speed in the relevant area.

It is believed that the combined effect of these phenomena will determine the thickness of the sludge layer that will accumulate inside the tank, but it was not possible to evaluate how much the sludge layer would accumulate based on the experimental results alone. However, it can be assumed that significant sludge accumulation occurred in areas where fluid flow was minimal and sludge formation nuclei easily accumulates.

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Keywords: Nuclear power plant decommissioning, Radioactive laundry liquid waste, Treatment of radioactive waste, Radioactive sludge

Silver and Copper Incorporated Mesoporous Silica for Hydrogen Isotope Separation via Chemical Affinity Quantum Sieving

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Tritium is radioactive isotope, emitting beta ray, released as tritiated water from nuclear power plants. Due to the danger of radioactive isotope, the appropriate separation of tritium is essentially carried out for environment and safety. Further, it is also promising material for energy production and research. The tritiated water can be treated by diverse techniques such as water distillation, cryogenic distillation, Girdler-sulfide process, and catalytic exchange. After treatment, it is more desirable to convert as gas phase for storage, comparing to liquid phase. However, achieving complete separation of hydrogen gases with very similar physical and chemical properties is significantly challenging. Thus, it is necessary to develop materials with effective separation properties in gas separation.

In this presentation, we present hydrogen isotope separation in the gas phase using modified mesoporous silica. Mesoporous silica is a form of silica that is characterized by its mesoporous structure possessing pores that range from 2 to 50 nm in diameter. This material can be functionalized to selectively capture and separate molecules having specific size and affinity. Here, the silver and copper incorporated mesoporous silica was synthesized to tailor a chemical affinity quantum sieving effect, thereby providing separation efficiency in D₂/H₂. The adsorption quantities of H₂ and D₂ were determined by sorption study, and the textural properties of each mesoporous silica were analyzed using N₂ physisorption. The selectivity (D₂/H₂) in diverse feed composition (1:1, 1:9, and 1:99 of D₂/H₂) was estimated by applying ideal adsorbed solution theory to predict the loading of the gas mixture on bare, Ag- and Cu-mesoporous silica based on their sorption study. Further, the performance of each mesoporous silica was evaluated in the breakthrough adsorption under 1:1 mixture of D₂ and H₂ at 77 K.

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Keywords: Hydrogen isotope, Tritium, Separation, Physisorption, Mesoporous silica, Chemical affinity quantum sieving

Development of a Corrosion Inhibitor to Reduce Secondary Waste in Heavy Water Reactor Core Systems

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Heavy water primary system decontamination technology is essential to reduce worker exposure and improve safety during maintenance and decommissioning of nuclear facilities. Advanced decontamination technology development aims to secure controlled decontamination technologies that can reduce the cost of radiation exposure and dramatically reduce the amount of secondary waste generated when decontaminating large equipment and large-area facilities.

We conducted a study to identify candidate corrosion inhibitors through the literature and analyze the degree of corrosion of carbon steel samples. Countries with advanced nuclear technology have developed chemical decontamination technology for the entire nuclear power generation system and applied it to the dismantling and maintenance of nuclear power plants. In the decontamination process, the corrosion oxide film must be removed. If the base metal is corroded by the decontaminant in this process, additional secondary waste is generated and treatment costs increase. Therefore, it is necessary to develop a corrosion inhibitor that inhibits the corrosion of the carbon steel base metal in the decontamination process to generate a secondary waste liquid that is favorable for waste reduction and treatment.

In this presentation, a study was conducted to analyze the extent of corrosion on a carbon steel base material and identify candidate materials for corrosion inhibition testing. Samples were analyzed using optical microscopy and EPMA analysis to determine the thickness of the corroded oxide film. EPMA analysis also allowed us to map the elemental distribution of the carbon steel corrosion layer, which we plan to quantify in the future. The candidate materials for organic-based corrosion inhibitor were also selected based on their inhibition mechanism; having high electronegative elements for coordinate covalent bonding at metal surface and hydrophobic nonpolar group for preventing access of corrosive substances. The selection of candidate materials for corrosion inhibition testing was based on the mechanism of the corrosion inhibitor. Organic-based corrosion inhibitors are adsorbed by donor-acceptor interactions between metal surfaces and highly electronegative elements. Corrosion can also be inhibited by arranging hydrophobic nonpolar groups on metal surfaces in the solution direction to prevent access of corrosive substances.

Keywords: PHWR, Carbon steel, Corrosion, Corrosion inhibitor, HyBRID

A Study on Double Arcing Monitoring Method Using Voltage Data During Plasma Arc Cutting of STS304 Plate

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As the decommissioning of domestic nuclear power plants (Gori Unit 1 and Wolseong Unit 1) becomes more visible, many research projects are being conducted to safely and economically decommissioning of domestic nuclear power plants (NPPs). After permanent shutdown, decommissioning of NNPs proceeds through decontamination, cutting of main equipment, waste disposal and site restoration stages. And various technologies are applied at each stage. In particular, remote cutting of neutron induced structures (RV, RVI, etc.) is a technology used in developed countries in the cutting stage, and remote cutting has been evaluated as a core technology for minimizing workers' radiation exposure. Generally, remote cutting technologies are divided into mechanical/thermal/electrical cutting. Among various thermal cutting technologies, plasma arc cutting (PAC) is more economical and easily to remote control than other cutting technologies, and is also effective in cutting STS304 plates.

PAC is a thermal cutting technology that melts the base material at the cutting area with a plasma arc heat source and removes melted material by blowing it out with cutting gas. The cutting quality depends on the stand-off distance and power (current), material thickness, cutting speed, etc., while double arcing will occur if the cutting conditions are not suitable. A monitoring system that can confirm double arcing during remote cutting is necessary because double arcing can reduce cutting quality, increase secondary waste (increase kerf and aerosol), and cause non-cutting.

In this study, we used an ultrahigh-speed camera equipped with a band-pass filter to capture clear arc shapes, and measured voltage waveforms with a data acquisition system. We studied a monitoring method that can confirm the occurrence of double arcing by synchronizing the obtained arc shape and voltage waveform, and the effects of double arcing on the STS304 plates. The results of this study are expected to be helpful in the development of the remote cutting process using plasma arc cutting when decommissioning of domestic NPPs.

Acknowledgements

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Keywords: Plasma arc cutting, Double arcing, Voltage, Monitoring, STS304, Cutting quality

Characteristics of Cutting Surface According to Cutting Direction and Laser Output During Laser-Oxygen Hybrid Cutting for Nuclear Power Plant Dismantling Application

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Thermal cutting processes that can be applied to dismantling nuclear power plants include oxygen cutting, plasma cutting, and laser cutting. According to the global trend, research projects are being carried out in various countries to upgrade laser cutting, and many studies are also being conducted in Korea with plans to apply laser cutting processes when dismantling nuclear power plants. However, with the current technology level of the laser cutting process, the maximum thickness that can be cut is limited to 250 mm.

Therefore, in this study, a laser-oxygen hybrid cutting process was implemented by adding a laser heat source to the oxygen cutting process that can cut carbon steel with a thickness of 250 mm or more (RV, beam, column, beam, etc.) when dismantling the nuclear power plant. This has the advantage of improving the cutting speed and reducing the cutting width Kerf compared to conventional oxygen cutting.

In this research, the laser-oxygen hybrid cutting process consisted of laser cutting to which Raycus' 8 kW Fiber Laser power source was applied and oxygen cutting to which hydrogen was applied with Fuel Gas.

The oxygen torch was placed perpendicular to the test piece, and the laser head was irradiated by tilting 35° to 70°. The effects of cutting directions on quality and performance were studied, and cutting paths were selected by comparing cutting results. Thereafter, it was confirmed that there is an optimal laser output power according to the cutting thickness by studying the effect on the cutting surface quality by changing only the laser output power under the same cutting conditions. The results of this study are expected to be helpful in the remote cutting process using laser-oxygen hybrid cutting when dismantling domestic nuclear power plants in the future.

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Keywords: Laser-oxy hybrid cutting, Oxy-fuel cutting, Laser cutting, Carbon steel

A Review on Characterization of High Temperature Processed Waste Form

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The primary purpose of high temperature process of radioactive waste is to satisfy the waste acceptance criteria and volume reduction.

The WAC offers the guideline of waste form fabrication process. The WAC is defined as quantitative or qualitative criteria specified by the regulatory body, or specified by and operator and approved by the regulatory body, for radioactive waste to be accepted by the operator of a repository for disposal, or by the operator of a storage facility for storage. The main objective of WAC is to protect staff and general public and environment by the containment of radioactive material, limit external radiation level, and prevent criticality. The WAC also offers systematic management of radioactive waste by standardization of waste management operations, facilitation waste tracking, ensure safe and effective operation of operating facilities, etc.

Since the high temperature process for radioactive waste is considered in many countries, lots of codes and standards are considered. In many WACs, compressive strength, thermal cycle stability, radiation exposure stability, free liquid, and leachability are evaluation to understand the effect of solidified form to the disposal facility.

In this paper, systematical review on waste form will be discussed. In addition, brief result of characterization of waste form will be compared.

Keywords: Waste acceptance criteria, High temperature, Heat treatment, Waste form, Solidification

A Review on off Gas Treatment in High Temperature Processing

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The thermal treatment of radioactive waste attracts great attention. The thermal treatment offers lots of advantages, such as significant volume reduction, hazard reduction, increase of disposal safety, etc. There are various thermal technologies to waste. The developed technologies are calcination, incineration, melting, molten salt oxidation, plasma, pyrolysis, synroc, vitrification, etc. The off-gas treatment system is widely applied in the technologies to increase the safety and operation efficiency.

The thermal treatment generates various by-product and pollutants during the process. The dust or fly ash are generated as a particulate from almost every radioactive waste. The treatment of PVC related components generates hydrogen chloride, which usually brings corrosion of facility. The treatment of rubber and spent resin generates sulfur oxide, SO_x . The treatment of nitrile rubber generates nitrogen oxide, NO_x . The incomplete combustion of radioactive waste usually generates carbon oxide, CO_x . The process temperature also affects the generation of off gas, such as NO_x and/or CO_x .

Various off gas treatment components are organized for the proper treatment of the previously mentioned materials. In this study systematical review on off gas treatment will be reported. Also, worldwide experiences and developed facility will be reported.

Keywords: Heat treatment, High temperature, Radioactive waste, Off gas treatment, Vitrification

Adsorption Characteristics of Aqueous Re(VII) Onto HDPy- and HDTMA-Modified Bentonite Under Various Conditions

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Tc-99 is considered as one of the major fission products in the context of disposal of spent nuclear fuel, due to the long half-life and chemical stability. In the atmospheric aqueous solutions, Tc is expected to exist in the form of TcO_4^- and thus is considered as an environmental concern according to its high solubility and mobility. Therefore, the development of an effective and economically viable adsorbent for aqueous Tc(VII) is imperative from the perspective of decontamination and remediation of contaminated environments. In this work, the adsorption behaviors of Re(VII), as a chemical surrogate of Tc(VII), onto the bentonites modified with two different organic cations such as hexadecyl pyridinium (HDPy) and hexadecyl trimethylammonium (HDTMA) were quantitatively analyzed and compared with each other.

For the sorption experiment, adsorbents were prepared by surface modification of bentonite. Before the modification, the initial bentonite was pre-treated with 1 M NaClO_4 and then reacted with HDPy or HDTMA. The modification process was performed at room temperature for 24 hours with various concentrations of organic cations, which were set to a range of 50-400% compared to the cation exchange capacity (CEC) of bentonite. After the reaction, the dried and crushed modified bentonites were filtered with the sieve with a mesh size of 63 μm . Aqueous Re(VII) solutions were prepared by dissolution of NH_4ReO_4 (Sigma-Aldrich) in deionized water with three different Re(VII) concentrations of 10^{-4}M , 10^{-5}M , and 10^{-6}M . After that, the modified bentonite and the aqueous Re(VII) solutions were mixed at a liquid-to-solid ratio of 1 g/L. Aliquots of the samples were extracted for quantification analysis with ICP-MS after syringe filtration (pore size: 45 μm) at reaction times of 10, 50, 100, and 500 minutes.

According to the results, a considerably fast adsorption reaction of Re(VII) onto all modified bentonites was observed, revealing exceptional sorption affinity of HDPy- and HDTMA-modified bentonites. For both organic cations, bentonites modified with the concentrations of organic cations ranging from 200 to 400% relative to the CEC of bentonite showed almost complete removal of aqueous Re(VII). For bentonites modified with lower concentrations of organic cations, the HDTMA presented a relatively larger sorption capacity than the HDPy. The result obtained through this study is expected to be referred to as a case study for the synthesis of cost-efficient and highly effective adsorbent material for highly mobile anionic radionuclides such as I^- and TcO_4^- .

Keywords: Adsorption, Technetium, Rhenium, HDPy, HDTMA, Bentonite

Concept Development for Performance Improvement of Evaluation and Classification System for NPP Decommissioning Soil Wastes

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Recently, BNS (Best System) developed a system for evaluation and classification of soil and concrete wastes generated from nuclear power plant decommissioning. It is composed of various modules for container loading, weight measurement, contamination evaluation, waste classification, stacking, storage and control. The contamination evaluation module of the system has two sub modules. One is for quick measurement with NaI (TI) detector and the other is for accurate measurement with HPGe detector. The container used at the system for wastes handling has capacity of 100 kg and made of stainless steel. According to the measurement result of Co-60 and Cs-137, the waste is classified as waste for disposal or waste for clearance. Performance of the system was demonstrated using RM (Reference Material) radiation source. This year, necessity of system improvement was suggested due to revised operation requirements. So, the system should show throughput of more than 1 ton/hr and Minimum Detectable Activity (MDA) of less than 0.01 Bq/g (1/10 of criteria for regulatory clearance) for Co-60 and Cs-137. And soil waste become main target of the system. For this, the container used for soil waste handling should have capacity of 200 kg. As a result, material for the container need to be changed from stainless steel to plastic or FRP (Fiber Reinforced Plastics). And large area detector should be introduced to the system to enhance processing speed of the system. Additionally, container storage rack and conveyor system should be modified to handle 200 kg capacity container. Finally, moving path of the container will be redesigned for enhanced throughput of the system. In this paper, concept development of the system was suggested and based on that, system development will be followed.

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Keywords: NPP decommissioning, Soil wastes, Waste classification, System improvement

A Study on the Decommissioning of Calandria in Wolsong Unit 1

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KHNP is conducting research to decommission Wolsong Unit 1 Calandria. Establishment of preparation and dismantlement processes, conceptual design of equipment and temporary radiation protection facilities, and waste management are being established. In particular, the ALARA plan is to be established by performing exposure dose evaluation for workers.

This study aims to deal with the methodology of evaluating exposure dose based on the calandria dismantling process. The preparation process consists of bringing in and installing tooling and devices, and removing interference facilities to secure work space. The main source term for the preparation process is the calandria structure itself and crud of feeders. In the case of the dismantlement process, a structure with a shape that changes according to the process was modeled as a radiation source.

It is intended to estimate the exposure dose by selecting the number of workers, time, and location required for each process in the radiation field evaluated according to the preparation and dismantlement process. In addition, it is also conducting an evaluation of the impact on dust generated by cutting operations and the human impact of C-14, H-3, which are specialized nuclides for heavy water reactors.

KHNP is conducting an exposure dose evaluation based on a process based on the preparation and dismantlement process for decommissioning Calandria through computation code analysis. If additional worker protection measures are deemed necessary through dose evaluation according to this methodology, the process is improved to prepare for the dismantling of worker safety priorities.

Keywords: Calandria, Wolsong unit 1, Decommissioning, Process, ALARA

Development of Scabbed Concrete Treatment Technology Using Plasma Treatment Technology

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Plasma melting technology uses electrical arc phenomena such as lightning to create high-temperature sparks of about 1,600 degrees or more to meet waste disposal requirements through treatment and reduction without distinguishing radioactive waste generated during nuclear power plant operation and dismantling according to physical characteristics. Decommissioning radioactive waste scabbed concrete occurs when polishing and cutting the contaminated structure surface to a certain depth. In this study, Scabbed concrete containing paint was confirmed for volume reduction and disposal safety using plasma treatment technology, and it is planned to be verified through continuous empirical tests.

Keywords: Plasma technology, Scabbed concrete, Melt, Arc

Review of NPP Decommissioning Requirements to Establish NPP Decommissioning Demonstration Plan

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The nuclear power plant (NPP) decommissioning market is expected to expand not only domestically but also overseas. Proven technologies must be applied to decommission NPP. This is based on Article 41-2, Paragraph 2 of the domestic ‘Enforcement Decree Of The Nuclear Safety Act’. Proven technology refers to technology that has verified that it can be applied in the field through demonstration. In other words, in order to carry out NPP decommissioning, verification must be done. Demonstration refers to reducing technological uncertainty and directly verifying services implemented in the field.

From a technology commercialization perspective, demonstration requires an approach based on technology readiness level (TRL) from a technology perspective and market readiness level (MRL) from a market perspective. The characteristics of demonstration also differ depending on the characteristics of each field. The demonstration in the field of nuclear energy is the demonstration of demand matching. This is to confirm the feasibility of the technology in the company’s required environment. In order to perform demonstration, a scenario must be derived by reflecting demonstration design considerations. After evaluating the derived scenario, an actual assessment is conducted using lab-based demonstration/virtual environment demonstration/real environment demonstration. What must be preceded by an actual assessment is confirming the consumer’s requirements.

In this study, the necessary environment and requirements of consumer’s to perform NPP decommissioning were reviewed. The domestic decommissioning procedure requirements management system presents decommissioning procedures, potential worker accidents, and worker requirements. In the case of foreign countries, it was confirmed that complex wide need, cost benefit, risk reduction, waste generation, operation, reliability and maintenance (RAM) improvement and quantitative measures were evaluated for the technology to be demonstrated. Also the requirements for demonstrating decommissioning need to a detailed review of actual decommissioning cases. Therefore, a comparison must be made between the requirements based on actual NPP decommissioning cases and the requirements derived from this research process. Afterwards, the empirical research approach proposed by the Ministry of Trade, Industry and Energy was applied. The empirical research approach proposed by the Ministry of Trade, Industry and Energy is to secure a track record over a certain period of time and performance under conditions similar to the actual environment in the final research stage at the TRL level 6 to 8. Through this, it will be possible to confirm the suitability of overseas technology for domestic application.

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Keywords: Decommissioning, Demonstration, Business requirement, KRID

Performance Evaluation in Actual Seawater to Verify the Feasibility of Extracting Dissolved Oceanic Resources

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In the ocean, there exist infinite resources, including certain metallic elements that can serve as potential energy sources. One of the methods for extracting these dissolved resources from seawater involves adsorption. This study discusses the results of experiments conducted in real seawater using a developed fiber-type adsorbent capable of extracting dissolved oceanic resources. The fiber-type adsorbent was deployed in seawater to adsorb the elemental resources. It was then retrieved after 2, 3, and 4 weeks for evaluation of its adsorption performance. The evaluation was carried out by dissolving the adsorbent in a strong acidic solution and calculating the adsorption amount per gram of adsorbent using ICP-MS. The results indicated that the adsorption performance was slightly lower than previously reported values. Nevertheless, it confirmed the feasibility of adsorbing and recovering dissolved resources from actual seawater.

Keywords: Adsorbent, Seawater, Fiber-type, Energy sources

Preliminary Study on the Safety Requirements Related to the Demonstration of Decommissioning Techniques

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Currently, Korea is planning to use various equipments and technologies for cutting, decontamination, compression, solidification, and packaging at decommissioning site of Kori unit 1 and Wolsung unit 1. In particular, Korea is considering to apply new technologies like inorganic acid decontamination, spent resin treatment technology not only to localize various decommissioning technology, but to meet the limit of 14,500 drums of the decommissioning waste per unit.

However, before the techniques applied to decommissioning, it is necessary to demonstrate the effectiveness and the safety of the techniques. Because unlike the industrial fields, the failure of the decommissioning technique in nuclear power plants can cause the spread or leakage of radioactive materials. In the 「Regulation on Technical Standards for Nuclear Reactor Facilities, Etc.」 is stated that the licensee shall apply proven technology to decommissioning and if the licensee apply new technology, he must provide reasonable evidence and prove its safety. In accordance with this approach, Nuclear Safety and Security Commission (NSSC) Notice No. 2021-24 can be applied to the decommissioning technology as one of a technical standard related with the demonstration of it. And it states 9 kinds of elements related to the radioactive waste management facilities and components like the management of radioactive effluents, prevention of contamination and overflow by radioactive materials, etc. But, they are mixed with the radioactive material considerations and industrial considerations, and these considerations are usually for the facilities, not equipments or techniques.

On the other hand, in the IAEA Safety Standards Series No. WS-G-2.1 Section 6.15 to 6.20, it recommended to evaluate 12 considerations for the decontamination technique and 7 considerations for the dismantling techniques. The Decommissioning Guide in Germany recommends to consider 3 conditions for radiation protection of decontamination techniques and 4 conditions for dismantling techniques.

Therefore, it is necessary to compare the safety requirements or recommendations related to the demonstration of decommissioning technology with the other countries to check there is something to learn from it.

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Keywords: Decommissioning, Decommissioning technology, Safety requirements, Demonstration, Regulation

Preliminary Design and Implement Plan of Kori Unit 1 Full System Decontamination Project

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KEPCO KPS is the contractor for the full system decontamination (FSD) of Kori Unit 1 and under preparation such as modification, lay out for equipment installation, setting up tie-in/out point for chemical injection and way to pressurize the system, of its successful performance. In this research, KPS introduced how KPS has designed and prepared for the FSD project and how will the chemical decontamination process be implemented.

As described in the previous research, chemical decontamination process is planned to be conducted for three cycles and each cycle is consisted of oxidation, reduction, decomposition, and purification. Oxidation and reduction process were conducted at 90°C. Chemical decomposition and purification process were conducted at 40°C due to the damage of IX by the heat. If the decontamination result does not meet the target DF and the dose rate, additional cycle can be conducted. Expected volume of process water for FSD is 200 m³.

Three systems have been designated as decontamination targets: reactor coolant system (RCS), residual heat removal system (RHRS), chemical volume control system (CVCS). For the steady flow rate, existed plant equipment such as reactor coolant pump (RCP) will be operated and modifications on some components will be conducted.

Due to the limited space for installation, decontamination equipment and other resources are distributed to three different places. KPS designed the layout of equipment installed inside the containment vessel. The layout contains the information of shielding for highly radiated equipment such as IX and filter skid.

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Keywords: Full system decontamination, Implement plan, Chemical decontamination, Plant modification, Equipment layout

Activated Carbon Waste Pretreatment System for Boron Containing Radioactive Waste Upcycling

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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, our group suggested the waste up-cycling process synthesizing B₄C neutron absorber using boric acid waste and activated carbon waste to innovatively reduce radioactive wastes. Radioactive activated carbon waste was utilized in off gas treatment system of NPP to capture nuclide such as I-131, C-14 and H-3. Activated carbon waste is treated as low-level radioactive waste and pre-treatment system for removing nuclide from the activated carbon waste is needed to use B₄C up-cycling process.

In this study, microwave treatment system is suggested to treat the activated carbon waste. Activated carbon waste was exposed to microwave for a few minutes and temperature of the waste was dramatically increased over 400°C. Nuclide in the activated carbon waste were selectively removed from the waste without massive production of secondary off gas waste.

Keywords: Microwave, Activated carbon waste, Radioactive, Up-cycling, B₄C

Glass-Ceramic Based Immobilization of By-Products Containing Co-60 From Thermochemical Treatment for Dry Active Waste

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Various dry active wastes (DAWs) have been accumulated in nuclear power plants since the DAWs are mostly combustible. KAERI has developed a thermochemical treatment process for the used decontamination paper as an operational waste to substitute for incineration process and to decontaminate radionuclides from the DAWs. The thermochemical process is composed of thermal decomposition in a closed vessel, chlorination of carbonated DAWs, separation of soluble chlorides captured in water by hydroxide precipitation, and immobilization of the precipitate. This study examined the third and fourth steps in the process to immobilize Co-60 by fabricating a stable wasteform. Precipitation behaviors were investigated in the chloride solution by adding 10 M KOH. It was shown that the precipitates were composed of $\text{Mg}(\text{OH})_2$ and $\text{Al}(\text{OH})_3$. Then, the glass-ceramic wasteform for the precipitates were produced by adding additive mixtures in which silica and boron oxide were blended with various ratios. The wasteform was evaluated in terms of volume reduction ratio, bulk density, compressive strength, and leachability.

Acknowledgements

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Keywords: Waste immobilization, Glass-ceramic composite, Dry active waste, Waste reduction

Study for Separation of Hydrogen Isotopes by Cation Exchanged Zeolite Beads

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The physicochemical similarities of hydrogen isotopes have made their separation a challenging task. Conventional methods such as cryogenic distillation, Girdler sulfide process, chromatography, and thermal cycling absorption have low separation factors and are energy-intensive. To overcome these limitations, research has focused on kinetic quantum sieving (KQS) and chemical affinity quantum sieving (CAQS) effects for selective separation of hydrogen isotopes. Porous materials such as metal-organic frameworks (MOF), covalent organic frameworks (COF), zeolites, carbon, and organic cages have been studied for hydrogen separation. In this study, we focus the enhancement for CAQS to provide the cations due to the chemical affinity between hydrogen isotope and unsaturated sites by cations in zeolite beads. Cation exchanged zeolite beads was synthesized with cobalt, copper, nickel, iron and silver in zeolite 4A beads. Synthesized cation exchanged zeolite was analyzed for the surface area and pore size in N₂ and adsorption behaviors of hydrogen isotopes (D₂/H₂) for various cation exchanged zeolite beads using BET at 77 K. The study predicts the D₂/H₂ adsorption selectivity based on the results obtained with BET. These hydrogen isotope adsorption results will provide a foundation for future processes for tritium separation.

Acknowledgements

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Keywords: Hydrogen, Isotope, Separation, Cation-exchanged, Zeolite beads

Preliminary Study for Site Remediation Strategies Focusing on the Distribution Coefficient and Hydrogeological Characteristics

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The operation of nuclear facilities involves the potential for on-site contamination of soil, primarily resulting from pipe leaks and other operational incidents. Globally, decommissioning process for commercial nuclear power plants have revealed huge-amounts of soil waste contaminated with Cs-137, Sr-90, Co-60, and H-3. For example, Connecticut Yankee in the United States produced approximately 52,800 ton of contaminated soil waste, constituting 10% of the total waste generated during its decommissioning. Environmental remediation costs associated with nuclear decommissioning in the US averaged \$60 million per unit, representing a significant 10% of the whole decommissioning expenses. Consequently, this study undertook a preliminary investigation to identify important factors for establishing a site remediation strategy based on radionuclide- and site-specific media- characteristics, focusing the efficiency enhancement for the environmental remediation.

The factors considered for this investigation were categorized into physical/environmental, socio-economic, technical, and management aspects. Physical/environmental factors contained the site characteristics, contamination levels, and environmental sensitivity, while socio-economic factors included the social concerns and economic costs. Technical and management factors included subcategories such as technical considerations, policy aspects, and management factors. Especially, technical factors were further subdivided to consider the site reuse potential, secondary waste generation by site remediation, remediation efficiency, and remediation time. Additionally, our study focused the key factors that facilitate the systematic planning for the site remediation, considering the distribution coefficient (K_d) and hydrogeological characteristics associated with each radionuclide in specific site conditions. Therefore, key factors in this study focus the geochemical characteristics of site media including the particle size distribution, chemical composition, organic and inorganic constituents, and soil moisture content. Moreover, the adsorption properties of site media were examined concerning the distribution coefficient (K_d) of radionuclides and their migration characteristics.

Furthermore, this study supported the development of a conceptual framework, containing the remediation strategies that incorporate the mobility of radionuclides, according to the site-specific media. This conceptual framework would necessitate the spatial analysis techniques involving the whole contamination surveys and radionuclide mobility modeling data. By integrating these key factors, the study provides the selection and simulation of optimal remediation methods, ultimately offering the estimated amounts of radioactive waste and its disposal costs. Therefore, these key factors offer foundational insights for designing the site remediation strategies according the site-specific information such as the distribution coefficient (K_d) and hydrogeological characteristics.

Acknowledgements

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Keywords: Decommissioning site, Environmental remediation, Establishing strategies, Radionuclides, Distribution coefficient, Hydrogeological characteristics

Adsorption Behavior of Radionuclides for Various Clays in Different pH, and Competition Cations and in Seawater-Infiltrated Groundwater and Seawater

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The mobility of radionuclides in the subsurface environment is governed by a interaction of radioactivity characteristics and geochemical conditions with adsorption reactions playing a critical role. This study investigates the characteristics and mechanisms of radionuclides adsorption on site media in viewpoint of nuclear safety, particularly focusing on the potential effect of seawater infiltration in coastal site near nuclear power plant. Seawater intrusion alters the chemistry in groundwater, including parameters such as pH, redox potential, and ionic strength, thereby affecting the behavior of radionuclides. To assess the safety of site near nuclear power plant and the environmental implications of nuclide leakage, this research conducted various experiments to evaluate the behavior of radionuclides in the subsurface environment.

High distribution coefficients (50-2,500 ml/g) were observed at 10 mg/L Co, with montmorillonite > hydrobiotite > illite > kaolinite. It decreased with competing cations (Ca^{2+}) and was found to decrease significantly by 90% with a decrease in pH to 4. It is believed that the adsorption capacity of cationic radionuclides decreases significantly as the clay mineral surface becomes less negatively charged. For Cs, the distribution coefficient (180-560 ml/g) was higher for montmorillonite > hydrobiotite > illite > kaolinite. Compared to Co, it was found to be less influenced by pH and more influenced by competing cations. For Sr, the distribution coefficient (100-380 ml/g) was higher in the order of hydrobiotite > montmorillonite > illite > kaolinite. Compared to Cs, it was found to be less affected by pH and also less affected by the effect of competing cations compared to Cs.

Seawater samples from Gampo and Uljin site near Nuclear Power Plant in Korea were analyzed to determine their chemical composition, which was subsequently used in adsorption experiments. Additionally, the seawater-infiltrated groundwater was synthesized in laboratory according to previous literature. The study focused on the adsorption and behavior of three key radionuclides such as cesium, strontium, and cobalt onto four low permeability media (clay minerals) such as kaolinite, illite, hydrobiotite, and montmorillonite known for their high adsorption capacity at a site of nuclear power plant. At concentrations of 5 and 10 mg/L, the adsorption coefficients followed the order of cobalt > cesium > strontium for each radionuclide. Notably, the distribution coefficient (K_d) values exhibited higher values in seawater-infiltrated groundwater environments compared to seawater with relatively high ionic strength. Cobalt exhibited a substantial adsorption coefficient, with a marked decrease in K_d values in seawater conditions due to elevated ionic strength. In contrast, cesium displayed less dependency on seawater compared to other radionuclides, suggesting distinct adsorption mechanisms, possibly involving fractured edge sites (FES) in clay. Strontium exhibited a significant reduction in adsorption in seawater compared to groundwater in all K_d sorption experiments.

The adsorption data of cobalt, cesium, and strontium on clay minerals in contact with seawater and seawater-infiltrated solutions offer valuable insights for assessing radioactive contamination of groundwater beneath coastal site near nuclear power plant sites. This research provides a foundation for enhancing the safety assessment protocols of nuclear power plant sites, considering the potential effects of seawater infiltration on radionuclide behavior in the subsurface environment.

Acknowledgements

This study was supported by the National Research Foundation of Korea (NRF) grant (No. 2022 M2D2A1A02063569) funded by the Ministry of Science and ICT.

Keywords: Radionuclides, Clay mineral, Distribution coefficient, Seawater, Seawater-infiltrated groundwater

Photo-Fenton Degradation of Bisphenol a by New Copper Based Metal-Organic Framework at Natural pH

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Bisphenol-A, also known as BPA, is commonly used as a building block for epoxy and polycarbonate plastics. However, it has been recently identified as a major source of water pollution due to its release into the water from plastic products. BPA-based resins can also contaminate the water with high concentrations of BPA, which can enter the water bodies through production units and wastewater discharge. Photocatalysis, particularly the photo-Fenton process, is an effective method for wastewater treatment and degrading pollutants. Titanium dioxide (TiO₂) is usually chosen based on its high photocatalytic properties and high performance. However, its wide band gap energy is a major issue for the photocatalytic process. This means that the catalyst can only exhibit high photocatalytic performance under UV-light irradiation and usually requires an acidic pH, which limits its use.

In order to address the aforementioned issues, a visible-light photoactive photo-Fenton reaction has been successfully developed to degrade bisphenol A at natural pH using H₂O₂. The process was highly efficient, achieving complete degradation of phenol in just three hours of visible light irradiation with Cu-MOF. This environmentally friendly Fenton process has the advantage of occurring at natural pH levels with the presence of H₂O₂, providing a new perspective for efficient degradation. The photocatalyst was characterized using single X-ray diffraction (SC-XRD), powder X-ray diffraction (PXRD), Fourier Transform Infrared Spectroscopy (FTIR), and UV-vis diffuse reflectance spectroscopy (DRS).

Keywords: Metal-organic framework, Methylene blue degradation, Photocatalyst

Direction of Development of Portable Air Purification Devices for Nuclear Power Plant Decommissioning

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In the dismantling of nuclear power plants, various forms of radioactive gaseous waste are generated when cutting concrete and metal structures. Large amounts of radioactive dust and aerosols generated during the cutting process of each structure can cause radiation exposure to the environment around the workplace and to the radiation exposure in the body of workers. When cutting structures, water is sprayed to reduce the generation of aerosols, so early saturation of the filter is expected due to radioactive aerosols and fine particles containing a large amount of moisture. A mobile air purification device is being developed to a fast and efficient air purifier that can be used for a long time operation to protect workers from radiation exposure in high radiation areas and to minimize the amount of secondary waste generated. In this paper, the direction for a new concept of unit technology that can achieve the development purpose is described.

Acknowledgements

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Keywords: Portable air purifier, Radioactive aerosol, NPP decommissioning, Gaseous wastes

Immobilization of Copper Hexacyanoferrate on Highly Porous Al₂O₃ Ceramic Foam for Remediation of Radioactive Cs Contaminated Water

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Copper hexacyanoferrate (Cu-HCF), which is a type of Prussian Blue analogue (PBA), possesses a specific lattice structure that allows it to selectively and effectively adsorb cesium with a high capacity. However, its powdery form presents difficulties in terms of recovery when introduced into aqueous environments, and its dispersion in water has the potential to impede sunlight penetration, possibly affecting aquatic ecosystems. To address this, sponge-type aluminum oxide, referred to as alumina foam (AF), was employed as a supporting material. The synthesis was achieved through a dip-coating method, involving the coating of aluminum oxide foam with copper oxide, followed by a reaction with potassium hexacyanoferrate (KHCF), resulting in the in-situ formation of Cu-HCF. Notably, Copper oxide remained chemically stable, which led to the application of 1, 3, 5-benzenetricarboxylic acid (H₃BTC) to facilitate its conversion into Cu-HCF. This was necessary to ensure the proper transformation of copper oxide into Cu-HCF on the AF in the presence of KHCF.

The synthesis of Cu-HCF from copper oxide using H₃BTC was verified through X-ray diffraction (XRD) analysis. The manufactured adsorbent material, referred to as AF@CuHCF, was characterized using Fourier-transform infrared spectroscopy (FTIR) and thermogravimetric analysis (TGA). These analyses revealed the presence of the characteristic C≡N bond at 2,100 cm⁻¹, confirming the existence of Cu-HCF within the AF@CuHCF, accounting for approximately 3.24% of its composition. AF@CuHCF exhibited a maximum adsorption capacity of 34.74 mg/g and demonstrated selective cesium adsorption even in the presence of competing ions such as Na⁺, K⁺, Mg²⁺, and Ca²⁺. Consequently, AF@CuHCF effectively validated its capabilities to selectively and efficiently adsorb cesium from Cs-contaminating wastewater.

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Keywords: Radioactive cesium, Copper hexacyanoferrate, Alumina foam, In-situ method

Analysis of Laser Scanning to 3D Mesh Conversion Methods for Digital Modeling of Nuclear Decommissioning Site

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As nuclear decommissioning ventures become increasingly complex, the role of digitalization in facilitating and enhancing these operations is becoming indispensable. This transition to a more digitized approach presents a myriad of advantages, including: augmented avenues for data acquisition, analysis, and visualization to bolster dismantling strategies; simulations in virtual environments for operator training; precise forecasting of future waste emergence, culminating in refined cost estimations; and more immersive decommissioning visualizations for both operators and external stakeholders. Salient benefits conferred by the integration of digital technologies in decommissioning encompass improved collaboration, enriched knowledge transfer, clarity regarding present technological constraints, insights into key influencing factors, clearer criteria for technology selection, and a profound understanding of the potential challenges and merits of a broader incorporation of digital tools in decommissioning endeavors. Of paramount importance is the opportunity presented for superior workforce training and safety measures, exemplified by ALARA-based planning.

Amidst the myriad facets of digital adoption, 3D modeling of nuclear facilities derived from laser-scanned point clouds stands out as a pivotal domain in the digitalization. The transformation of intricate point cloud data into a comprehensible 3D mesh remains the crux of this paper. The process of mesh generation, despite being simpler than its counterpart of converting to a 3D solid model, is crucial for multiple reasons. The resultant 3D mesh offers an enhanced visual representation compared to a sparse point cloud, paving the way for improved spatial perception. Furthermore, it serves as a rudimentary tool for approximating component volumes and the ensuing waste, thereby playing an instrumental role in waste manipulation strategies, notably in collision detection.

This paper delves deep into the nuances of mesh generation, conducting a parametric study of mesh conversion algorithms, including down-sampling rates. Through this rigorous examination, we endeavor to shed light on optimal methodologies, hoping to catalyze advancements in the digitalization of nuclear decommissioning processes.

Acknowledgements

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Keywords: Nuclear decommissioning, Digitalization, Point cloud, 3D mesh, Laser scanning

A Study on the Optimization of Worker Exposure and Decommissioning Costs Through Scenario-Based Evaluation During Steam Generator Dismantling

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In South Korea, the replacement of steam generators began with Kori Unit 1 in 1995, and to date, 20 steam generators have been replaced and are currently stored in intermediate storage facilities. In the future, additional decommissioned steam generators may arise due to measures like the extension of the lifespan of nuclear power plants. In Korea, technological development for dismantling steam generators is underway, and there is no track record of actual dismantling. Although the replaced decommissioned steam generators are stored in intermediate facilities, for site recycling purposes, steam generators, which have relatively lower radiation doses compared to reactor heads and other primary equipment, should be prioritized for dismantling.

While there are various specifications for steam generators, those dismantled and stored domestically are of the Recirculation Type. They can be classified into three types: the Westinghouse type WH-51 used in Kori Unit 1, the Fra-51B used in Han-ul Units 1 and 2, and the OPR-1000 used in Han-ul Units 3 and 4. The quantity of U-Tubes varies depending on the specification, but the radiation is concentrated in the primary side components, the U-Tube and Chamber. Since the parts related to the secondary side are not contaminated, they can be disposed of independently after classification. To dismantle a steam generator, it is of utmost importance to first create a scenario regarding where and how the dismantling will take place. Through the analysis of the advantages and disadvantages of each scenario, the optimal timing, location, and cutting method for dismantling should be researched. Furthermore, based on those findings, the best scenario should be derived through an analysis of worker radiation exposure and dismantling costs.

To achieve this, a 3D simulation software developed by Cyclelife Digital Solutions under the French EDF was utilized to conduct simulations based on different dismantling schedules and methods. As a result, the optimal scenario for dismantling the steam generator was derived.

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Keywords: Decommissioning, DEMplus, Steam generator

Review of Wolsong Unit 1 Decommissioning Strategy Evaluation Plan Reflecting Decision-Making Technique

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Wolsong Unit 1, a domestic heavy water reactor nuclear power plant, was permanently shut down in December 2019. Accordingly, Wolsong Unit 1 plans to prepare a Final Decommissioning Plan (FDP), submit it to the government by 2024, receive approval for decommissioning, and begin full-scale decommissioning. One of the important tasks in the decommissioning of Wolsong Unit 1 is to determine the decommissioning strategy. It is necessary to decide on a decommissioning strategy considering various factors and variables, secure the technical background, and justify it.

The selection of a decommissioning strategy is best achieved through the use of formal decision-making assistance techniques, such as considerations related to influencing factors. It is very important to understand the basic decommissioning strategy alternatives and whether sufficient consideration has been given to situations where only a single unit is permanently shut down in a multi-unit site like Wolsong Unit 1, while the remaining units are in normal operation.

As a process for selecting a decommissioning strategy, first, all considerations that could potentially affect decommissioning presented in the KINS Decommissioning Safety Review Guidelines were synthesized, influencing factors to be used in the decision-making process were determined, and the concept was defined. In order to select the most appropriate decommissioning strategy by considering various evaluation attributes of possible decommissioning alternatives (immediate dismantling and delayed dismantling), the Wolsong Unit 1 decommissioning strategy was evaluated by reflecting the AHP decision-making technique.

Keywords: Decommissioning strategy, Influencing factor, AHP, Wolsong unit 1

Evaluation of Sintering Characteristics for Radioactive Precipitates Generated During Waste Treatment Processes

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Various types of radioactive liquid and solid wastes are generated during the operation and decommissioning of nuclear power plants. To remove radionuclides Co-60, Cs-137 etc. from a liquid waste, the ion-exchange process based on organic resins has been commonly used for the operation of nuclear facilities. Due to the considerations for the final disposal of process end-product, other treatment methods such as adsorption, precipitation using some inorganic materials have been suggested to prepare for large amounts of waste during decommissioning. This study evaluated sintering characteristics for radioactive precipitates generated during the liquid waste treatment process. The volume reduction efficiency and compressive strength of sintered pellets were the major parameters for the evaluation.

Major components of a simulated precipitate were some coagulated (oxy) hydroxides containing light elements, such as Si, Al, Mg, Ca, and zeolite particles. Green pellets compressed to around 100 MPa were heated at a range of 750~850°C to synthesize sintered pellets. It was observed that the volume reduction percentages were higher than 50% in the appropriate sintering conditions. The volume reduction was caused by the reduction of void space between particles, which is an evidence of partial glassification and ceramization of the precipitates. This result can also be attributed to conversion reactions of zeolite particles into other minerals. The compressive strength ranged from 6 to 19 MPa. These results also showed a significant correlation with the volume reduction of sintered body.

Although our lab-scale experiments showed many benefits of sintering for the precipitates, optimized conditions are needed for large-scale practical applications. Evaluation of sintering characteristics as a function of pellet size and further testing will be conducted in the future.

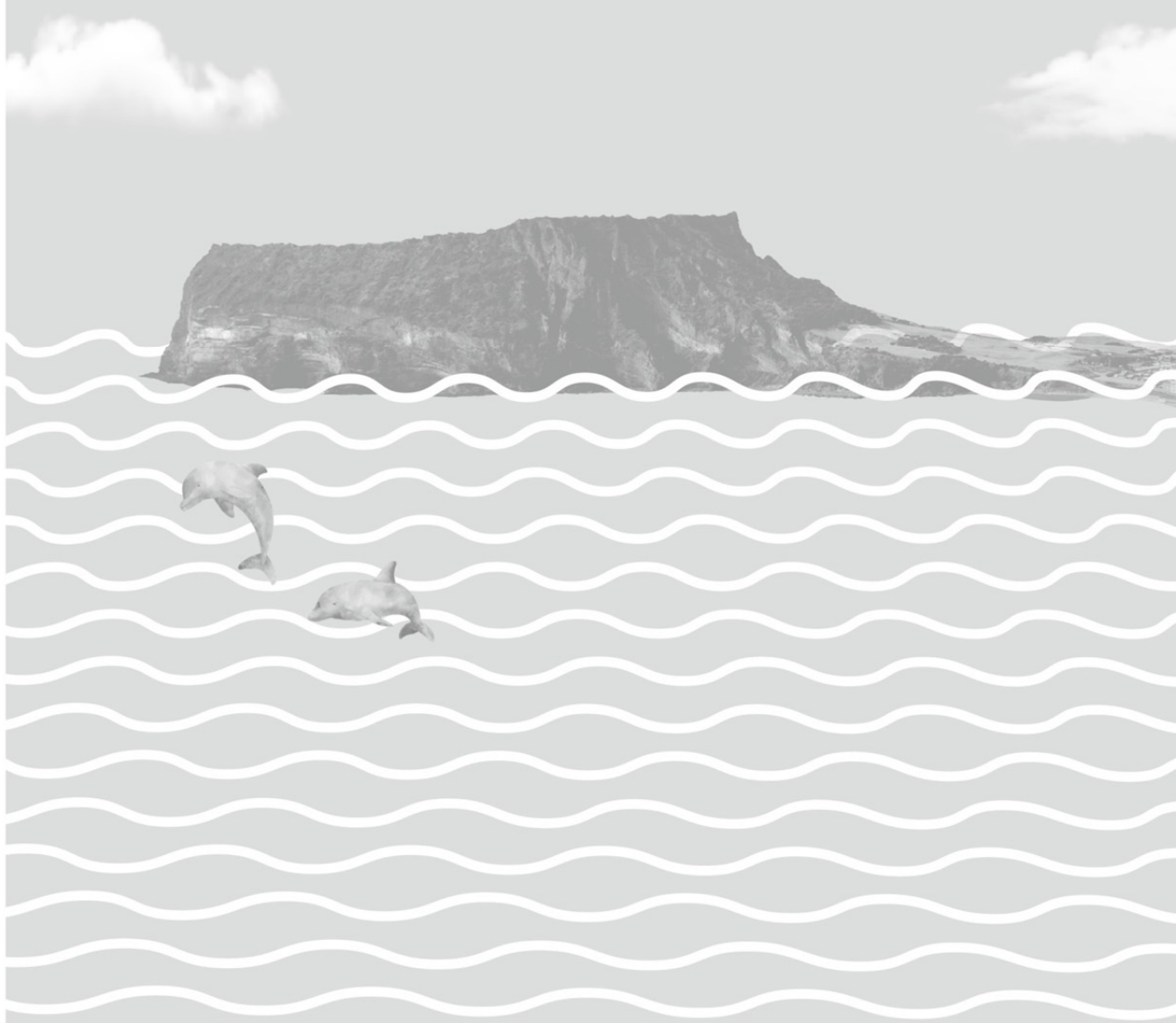
Acknowledgements

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Keywords: Decommissioning waste, Treatment, Stabilization, Radionuclide, Sintering

6분과

방사선환경 및 안전 (ORAL)



A Feasibility Study on the Gamma-ray Spectroscopy Using the CsPbBr₃ Perovskite Nanocrystal-Coated Scintillator to Detect Radionuclides With Higher Light Yield

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The inorganic scintillator used in gamma spectroscopy must have good efficiency in converting the kinetic energy of charged particles into light as well as high light output and high light detection efficiency. Accordingly, various studies have been conducted to enhance the net-efficiency. One way to improve the light yield has been studied by coating scintillators with various nanoparticles, so that the scintillation light can undergo resonance on surface between scintillators and nanoparticles resulting in higher light yield.

In this study, an inorganic scintillator coated with CsPbBr₃ perovskite nanocrystals using dip coating technique was proposed to improve scintillation light yield. The experiment was carried out by measuring scintillation light output, as the result of interaction between inorganic scintillator coated with CsPbBr₃ perovskite nanocrystals and gamma-ray emitted from Cs-137 gamma source. The experimental results show that the channel corresponding to 662 keV full energy peak in the Cs-137 spectrum shifted to the right by 14.37%.

Further study will be conducted to investigate the detailed relationships between the scintillation light yield and the characteristics of coated perovskite nanoparticles, such as diameter of nanoparticles, coated area ratio and width of coated region.

Keywords: Gamma-ray spectroscopy, Perovskite nanocrystal, Inorganic scintillator, Resonance, Radiation detection

Estimation of Naturally Occurring Radionuclides and the Annual Effective Dose by Drinking Groundwater of Jeju Island, Korea

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This study presents distribution of naturally occurring radioactive materials in groundwater in Jeju island. Radon (^{222}Rn) and potassium (^{40}K) concentrations were performed by using Liquid Scintillation Counter and Ion Chromatograph respectively. In addition, the activities of uranium and thorium nuclides were analyzed by Inductively Coupled Plasma Mass Spectroscopy. Groundwater samples were collected from 9 sites of water intake facilities for wide area supply in Jeju island from September 2022 to September 2023. The ^{40}K concentrations of groundwater ranged between 0.050 and 0.400 $\text{Bq}\cdot\text{L}^{-1}$. The radon concentrations in groundwater were in the range of 0 to 60 $\text{Bq}\cdot\text{L}^{-1}$, and there was no groundwater exceeding the range of 148 $\text{Bq}\cdot\text{L}^{-1}$ proposed by the US EPA. The distribution of uranium and thorium in groundwater varied from 0 to 500 $\text{ng}\cdot\text{L}^{-1}$ and 0 to 2.4 $\text{ng}\cdot\text{L}^{-1}$, respectively. The concentrations of uranium did not exceed 30 $\mu\text{g}\cdot\text{L}^{-1}$, thresholds indicated by the US EPA. By analyzing the concentrations of ^{40}K , ^{222}Rn , ^{238}U and ^{232}Th , the annual effective dose of residents can be assessed. The evaluated residents' effective dose from natural radionuclides due to intake of drinking water is less than the recommended value of 100 $\mu\text{Sv}\cdot\text{y}^{-1}$. Consequently, this study indicates that the cancer risks of the residents in Jeju island from naturally occurring radioactive materials ingested with water is insignificant.

Keywords: ^{222}Rn , ^{40}K , Uranium, Thorium, Ground water, Jeju

Development Status of Nuclear Power Plant Resident Exposure Dose Assessment Program Reflecting the Latest Regulatory Requirements

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To construct and operate nuclear power plants (NPPs), it is mandatory to submit a radiation environmental impact assessment report in accordance with Article 10 and Article 20 of the Nuclear Safety Act. Additionally, in compliance with Article 136 of the Enforcement Regulations of the same law, KHNP (Korea Hydro & Nuclear Power) annually assesses radiation environmental effects and publishes the results for operating NPPs. Furthermore, since the legalization of emission plans submission in 2015, KHNP has been submitting emission plans for individual NPPs, starting with the Shin-Hanul 1 and 2 units in 2018. These emission plans specify the emission quantities that meet the dose criteria specified by the Nuclear Safety and Security Commission. Before 2002, KHNP used programs developed in the United States, such as GASPARD and LADTAP, for nearby radiation environmental impact assessments. Since then, KHNP has been using K-DOSE60, developed internally. K-DOSE60 incorporates environmental transport analysis models in line with U.S. regulatory guidance Regulatory Guide 1.109 and dose assessment models reflecting ICRP-60 recommendations. K-DOSE60 is a stand-alone program installed on individual user PCs, making it difficult to manage comprehensively when program revisions are needed. Additionally, during the preparation of emission plans and the licensing phase, improvements to K-DOSE60's dose assessment methodology were identified. Furthermore, in 2022, regulatory guidelines regarding resident dose assessments were revised, leading to additional improvement requirements. Currently, E-DOSE60, being developed by KHNP, is a network-based program allowing for integrated configuration management within the KHNP network. E-DOSE60 is expected to be developed while incorporating the identified improvements from K-DOSE60, in response to emission plan licensing and regulatory guideline revisions. Key improvements include revisions to dose assessment methodologies for H-13 and C-14 following IAEA TRS-472, expansion of dose assessment points, and changes in socio-environmental factors. Furthermore, data such as site meteorological information and releases of radioactive substances in liquid and gaseous forms can be linked through a network, reducing the potential for human errors caused by manual data entry. Ultimately, E-DOSE60 is expected to optimize resident exposure dose assessment and enhance public trust in NPP operation.

Keywords: Nuclear power plant, Resident exposure dose assessment, K-DOSE60, E-DOSE60

Fault Detection and Diagnosis (FDD) for Heating, Ventilation and Air Conditioning (HVAC) Systems: A Review of Current Research Trends and Insight for Nuclear Fuel Cycle (NFC) Facilities

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In the nuclear fuel cycle (NFC) facilities, the failure of Heating Ventilation and Air Conditioning (HVAC) system starts with minor component failures and can escalate to affecting the entire system, ultimately resulting in radiological consequences to workers. In the field of air-conditioning and refrigerating engineering, the fault detection and diagnosis (FDD) of HVAC systems have been studied since faults occurring in improper routine operations and poor preventive maintenance of HVAC systems result in excessive energy consumption. This paper aims to provide a systematic review of existing FDD methods for HVAC systems therefore explore its potential application in nuclear field. For this goal, typical faults and FDD methods are investigated.

The commonly occurring faults of HVAC are identified through various literature including publications from International Energy Agency (IEA) and American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE). However, most literature does not explicitly addresses anomalies related to pressure, even though in nuclear facilities, abnormal pressure condition need to be carefully managed, particularly for maintaining radiological contamination differently within each zone.

To build simulation model for FDD, the whole-building energy system modeling is needed because HVAC systems are major contributors to the whole building's energy and thermal comfort, keeping the desired environment for occupants and other purposes. The whole-building energy modeling can be grouped into three categories: physics-based modeling (i.e., white-box models), hybrid modeling (i.e., grey-box models), and data-driven modeling (i.e., black-box models).

To create a white-box FDD model, specialized tools such as EnergyPlus for modeling can be used. The EnergyPlus is open source program developed by US-DOE, and features heat balance calculation, enabling the dynamic simulation in transient state by heat balance calculation. The physics based modeling has the advantage of explaining clear cause-and-effect relationships between inputs and outputs based on heat and mass transfer equations, while creating accurate models requires time and effort. Creating a black-box FDD model requires a sufficient quantity and diverse types of operational data for machine learning. Since operation data for HVAC systems in existing nuclear cycle facilities are not fully available, so efforts to establish a monitoring system enabling the collection, storage, and management of sensor data indicating the status of HVAC systems and buildings should be prioritized. Once operational data are available, well-known machine learning methods such as linear regression, support vector machines, random forests, artificial neural networks, and recurrent neural networks (RNNs) can be used to classify and diagnose failures. The challenge with black-box models is the lack of access to failure data from operating facilities. To address this, one can consider developing black-box models using reference failure data provided by IEA or ASHRAE.

Given the unavailability of operation data from the operating NFC facilities, there is a need for a short to medium-term plan for the development of a physics-based FDD model. Additionally, the development of a monitoring system to gather useful operation data is essential, which could serve both as a means to validate the physics-based model and as a potential foundation for building data-driven model in the long term.

Acknowledgements

This work was supported by the Korea Atomic Energy Research Institute (KAERI) grant funded by the Korea government (MSIT) (No. 524441-23).

Keywords: HVAC, Fault detection and diagnosis (FDD), Nuclear fuel cycle facility, Typical fault type, Diagnostic algorithm, FDD modeling, Physics-based modeling, Data-driven modeling

Analysis of Characteristics for Abnormal Releases of Radioactive Materials Reported From the U.S. and Korean PWRs

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With an ultimate view to identifying abnormal releases of radioactive materials, a set of liquid and gaseous effluent data including unplanned or uncontrolled releases annually reported from the U.S. and Korean nuclear power plants were systematically analyzed.

With the use of 21 years' worth of annual discharge data for 7 radionuclide groups and 24 individual radionuclides, taken from a combined total of 1,610 reactor-years (RYs) covering 62 units of US Pressurized Water Reactors (PWRs) and 22 units of Korean PWRs, three novel formulas for estimating events were employed to calculate characteristic values. Applying these characteristic values derived from the event estimation formulas to events that transpired during 699 RYs in operational US PWRs revealed an enhanced predictive accuracy for abnormal events when considering individual radionuclides, as opposed to grouping them by radionuclide groups. This effect was particularly pronounced for specific events such as leaks caused by problems in Gas Decay Tanks, leaks in Steam Generator Power Operated Relief Valves, fuel defects, and leaks during spent nuclear fuel processing.

In the case of Korean PWRs, fuel defects were identified as the primary events related to radioactive effluent releases. The methodologies and characteristic values derived from this study were applied to these events. The event estimation rate was lower in Korean compared to US PWRs, which can be attributed to the lower frequency of event occurrences in Korean PWRs (30 RYs) compared to the US.

The approach proposed in this study may contribute to develop a methodology to identify implicit abnormal release data and correlate them with specific operational occurrences or events, which could improve the conventional practice of simply recording and reporting radioactive discharge data.

Acknowledgements

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Keywords: Pressurized water reactor, Radioactive effluent, Abnormal event, Gas decay tank, PORV, Fuel defect

Directionality Performance Evaluation of a Multi-Channel Radiation Detector for Unmanned Aerial Vehicles

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If radioactive plumes are released outside due to loss of containment building integrity during a nuclear power plant accident, these materials might travel with the wind, affecting both the surrounding environment and neighboring countries. In China, most nuclear power plants are located on the eastern coast. Consequently, a radioactive plume generated during an accident could negatively impact even the western part of the Korean Peninsula due to westerly winds. To detect such problems early, respond quickly, and protect residents, a system that can monitor aerial radiation under normal conditions is needed. Additionally, a detection system that can operate in real-time in an emergencies conditions is required. The current method for aerial radiation measurement takes environmental radiation data from a monitoring post 1.5 m above the ground and converts it to altitude. To measure actual aerial radiation, an expansive area is surveyed by aircraft. However, this approach is both time-consuming and expensive. Thus, to monitor radioactive plumes influenced by environmental factors like wind, we need a radiation detector that can gauge both radioactivity and directionality.

In this study, we developed a radiation detector capable of assessing both the radioactivity and directionality of a radioactive plume and conducted its performance evaluation. We miniaturized the radiation detector using a CZT (Cadmium Zinc Telluride) sensor, enabling its mounting on unmanned aerial vehicles like drones. It is configured with multi-channels to measure directionality of a radioactive plumes.

For performance evaluation, we positioned two-channel CZT sensors at 90 degrees and measured the energy spectrum for angle and distance using a disk-type radioactive isotope. Using this method, we compared and analyzed the directionality performance of the multi-channel radiation detector. We also confirmed its capability to discern specific radioactivity information and nuclide types in actual radioactive plumes. Our future research direction involves mounting the multi-channel radiation detector on a drone. We aim to gather actual aerial radiation data from sensors positioned in various directions.

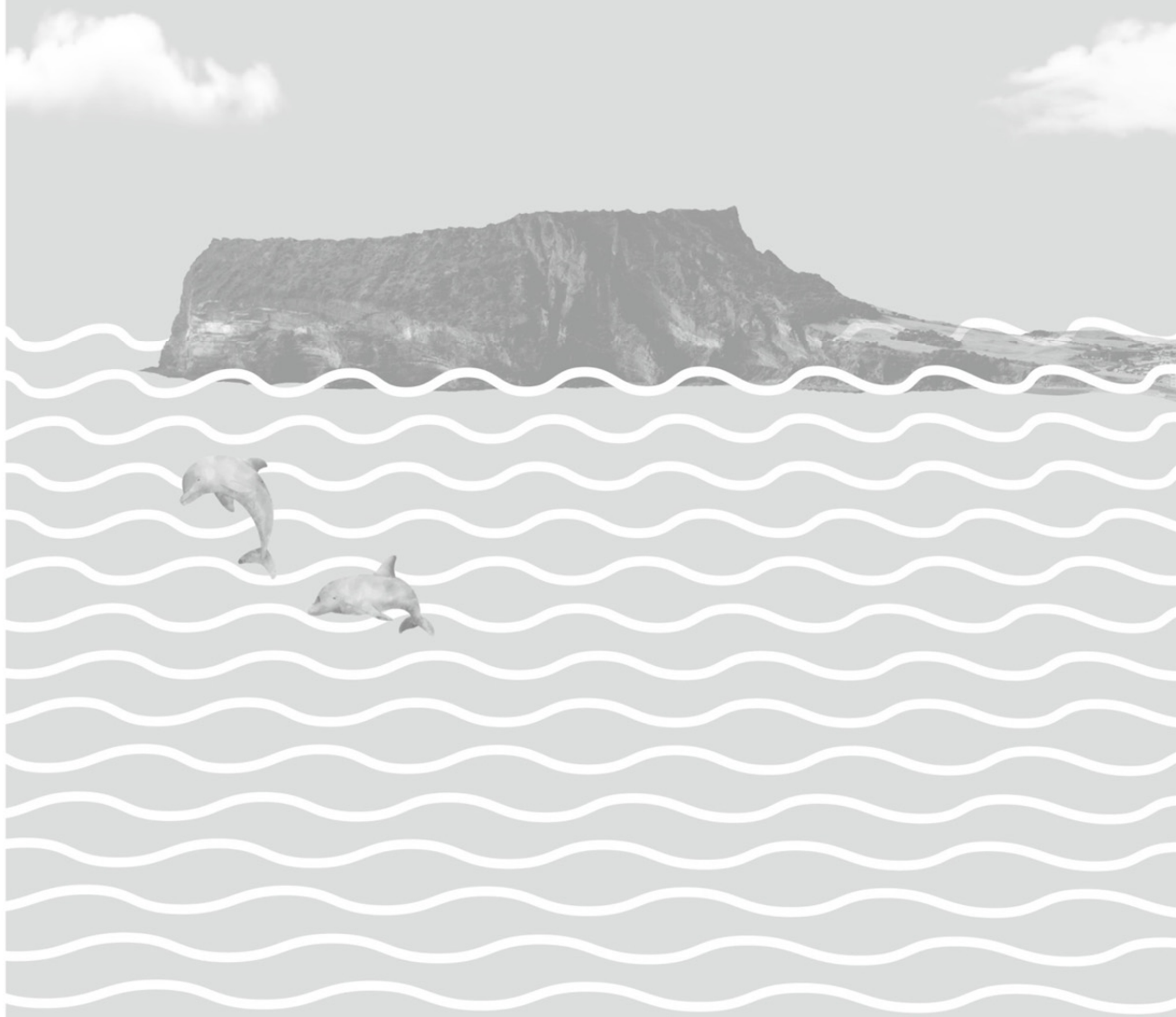
Acknowledgements

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Keywords: Aerial radiation, Radioactive plume, Multi-channel radiation detector, Directionality CZT (cadmium zinc telluride) sensor

6분과

방사선환경 및 안전 (POSTER)



Analysis of Site Hydraulic Characteristics for Combustible Radioactive Waste Treatment Facility (CRWTF) and Groundwater Modeling Validation

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To secure approval for a decommissioning plan in Korea, it is essential to evaluate contamination dispersion through groundwater during the decommissioning process. To achieve this, licensees must assess the groundwater characteristics of the facility's site and subsequently develop a groundwater flow model. It is worth noting that Combustible Radioactive Waste Treatment Facility (CRWTF) is characterized by their simplicity and absence of liquid radioactive waste generation. Given these facility characteristics, the groundwater flow model for CRWTF utilizes data from neighboring facilities, with the feasibility of using reference data substantiated through comparative analysis involving groundwater characteristic testing and on-site modeling.

To enable a comparison between the actual site's groundwater characteristics and the referenced modeling, two types of hydraulic constant characterization tests were conducted. First, hydraulic conductivity was determined through long-term pumping and recovery tests. The 'Theis' and 'Cooper-Jacob' equations, along with the 'Theis recovery' equation, were applied to calculate hydraulic conductivity, and the final result adopted the average of the calculated values. Secondly, a groundwater flow test was conducted to confirm the alignment between the main flow direction of the referenced model and the groundwater flow in the CRWTF, utilizing the particle tracking technique.

The evaluation of hydraulic conductivity from the hydraulic constant test revealed that the measured value at the actual site was approximately 1.84 times higher than the modeled value. This variance is considered valid, taking into consideration the modeling's calibration range and the fact that measurements were taken during a period characterized by wet conditions. Furthermore, a close correspondence was observed between the groundwater flow direction in the reference model (ranging from 90° to 170°) and the facility's actual flow direction (ranging from 78° to 95°).

The results of reference data for the CRWTF, based on the nearby facility's model, were validated through the hydraulic properties test. Consequently, the modeling data can be employed for the demolition plan of CRWTF. It is also anticipated that these comparative analysis methods will be instrumental in shaping the groundwater investigation plans for facilities with characteristics similar to CRWTF.

Acknowledgements

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Keywords: Combustible radioactive waste treatment facility, Site characteristic evaluation, Groundwater modeling validation, Groundwater character

Study on the Spectrum Features of the CdZnTe Detector Utilizing in Nuclear Material Safeguards

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Detectors utilized for nuclear material safeguards have been using scintillation detectors which are inexpensive and highly portable, and electrically cooled germanium detectors which are expensive but have excellent energy resolution. However, recently IAEA, the only international inspectorate of nuclear material safeguards for the globe, have replaced the existing scintillation detector and electrically cooled germanium detector with a CdZnTe detector owing to the improved performance of room-temperature semiconductors significantly.

In this paper, we will examine the spectrum features of the CdZnTe detector such as spectrum shape, energy resolution, and efficiency in the energy region of interest, which are the important characteristics for measuring Uranium enrichment. For this purpose, it would be conducted to compare its spectrum features using CdZnTe, NaI, HPGe detectors. The main energies of interest include 185.7 keV and 1,001 keV, which are the decay energies of uranium 235 and uranium 238.

The results of this study will provide a better understanding of the spectral features of various detectors used in uranium enrichment analysis, and are expected to be used as basic data for future related software development.

Acknowledgements

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Keywords: CdZnTe detector, Uranium enrichment, IAEA safeguards, Nuclear material, Spectral features

A Study on the Influence of Uranium Intake by Drinking Water on In-Vitro Bioassays

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Radiation workers, especially those dealing with Uranium isotopes, can potentially intake Uranium-containing materials through their respiratory and digestive systems. According to the “Regulations on the Measurement and Calculation of Internal Exposure” from Nuclear Safety and Security Commission (NSSC), those who intend to work in or enter the nuclear facilities with a risk of exceeding 2 mSv exposure per year should be examined the internal exposure. However, when it comes to in-vitro bioassay, Uranium intake through drinking water can affect the quantitative analysis. The International Commission on Radiological Protection (ICRP) reported in ICRP Publication 23 (Report on the Task Group on Reference Man) that the reference man excretes Uranium in the urine (0.05-0.5 µg/day) and feces (1.4-1.8 µg/day). Korea Atomic Energy Research Institute (KAERI) set the 90.5 ng/day as the ²³⁸U background of workers handling Uranium based on the daily Uranium intake of Koreans.

In this research, we examined the possible effects of Uranium in drinking water on internal exposure by analyzing the concentration of Uranium in bottled waters from various water sources sold in the domestic market and a water from the water purifier. The ²³⁸U concentration results of analyzing 11 bottled waters and 1 purified water, were ranged from 0 to 10.2 µg/L. All the results were satisfied the standard of 30 µg/L according to “Regulations for Drinking Water Quality Standards and Inspection” enacted by the Ministry of Environment. However, various concentrations were shown depending on the water sources. Assuming that these concentrations of water are consumed by drinking 1 L per day, the internal dose assessment result is 0 to 0.94 mSv. On the other hand, if it is assumed to be inhaled, it can be an overestimated because the dose coefficient of inhalation, Type M is higher than that of ingestion, $f_1=0.02$ which are the values recommended by ICRP Publication 78 (Individual Monitoring for Internal Exposure of Workers) when the Uranium compound is unspecified. In case of two workers at KAERI, the daily excretion of urine was 151 and 120 ng/day respectively in the first quarter monitoring. However after changing the kind of drinking water in the second quarter monitoring, it dropped to 17.4 and 15.4 ng/day respectively.

Through this study, it is confirmed that the Uranium background in urine can be analyzed differently depending on the kind of drinking water consumed by each worker. Depending on the Uranium concentration of drinking water, the internal exposure dose assessment can be overestimated or underestimated. Therefore, the Uranium concentration and intake amount according to the kind of drinking water should be considered for in-vitro bioassays of Uranium handlers. Furthermore, if necessary, the Uranium isotope ratio analysis in urine and the handling information should be comprehensively considered.

In addition, in order to exclude the effect of intake through the digestive system, replacing the kind of drinking water can be considered. The additional analysis such as in-vivo bioassay and 24 hours urine analysis rather than spot samples can be also recommended.

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Keywords: In-vitro bioassay, Urine, Uranium, Concentration measurement, ICP-MS, Drinking water

Analysis of the Ocean Transfer Evaluation Methodology for the Overseas Maritime Transport Risk Assessment Codes

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The demand for transportation is increasing due to the continuous generation of radioactive wastes. Especially, considering the geographical characteristics of Korea and the location characteristics of nuclear facilities, the demand for maritime transportation is expected to increase. If a sinking accident happens during maritime transportation, radioactive materials can be released into the ocean from radioactive waste transportation containers. Radioactive materials can spread through the ocean currents and have radiological effects on humans. The effect on humans is proportional to the concentration of radioactive materials in the ocean compartment. In order to calculate the concentration of radioactive materials that constantly flow along the ocean current, it is necessary to divide the wide ocean into appropriate compartments and express the transfer processes of radioactive materials between the compartments. Accordingly, this study analyzed various ocean transfer evaluation methodologies of overseas maritime transportation risk codes. MARINRAD, POSEIDON, and LAMER codes were selected to analyze the maritime transfer evaluation methodology. MARINRAD divided the ocean into two types of compartments that water and sediment compartments. And it was assumed that radionuclides are transferred from water to water or from water to sediment. Advection, diffusion, and sedimentation were established as transfer process for radionuclides between compartments. MARINRAD use transfer parameters to evaluate transfer processes by advection, diffusion, and sedimentation. Transfer parameters were affected by flow rate, sedimentation rate, sediment porosity, and etc. POSEIDON also divided the ocean into two types that water and sediment compartment, each compartments was detailedly divided into three vertical sub-compartment. Advection, diffusion, resuspension, sedimentation, and bioturbation were established as transport processes for radionuclides between compartments. POSEIDON also used transfer parameters for evaluating advection, diffusion, resuspension, sedimentation, and bioturbation. Transfer parameters were affected by suspended sediment rates, sedimentation rates, vertical diffusion coefficients, bioturbation factors, porosity, and etc. LAMER only considered the water compartment. It divided the water compartment into vertical detailed compartments. Diffusion, advection and sedimentation were established as the nuclide transfer processes between the compartments. To evaluated the transfer processes of nuclides for diffusion and advection, LAMER calculated the probability with generating random position vectors for radionuclides' locations rather than deterministic methods such as MARINRAD's transfer parameters or POSEIDON's transfer rates to evaluate transfer processes. The results of this study can be used as a basis for developing radioactive materials' ocean transfer evaluation model.

Acknowledgements

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Keywords: Maritime transportation, Transfer evaluation, MARINRAD, POSEIDON, LAMER

Derivation of Weighting Factor for Dose Coefficient Based on ICRP 107 Radionuclide Data

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When the parent radionuclide decays, the progeny radionuclide is produced. Accordingly, the dose contribution of the progeny radionuclide should be considered when assessing dose. For this purpose, European Commission (EC) and International Atomic Energy Agency (IAEA) provide weighting factors for dose coefficient. However, these weighting factors have a limitation that does not reflect the latest nuclide data. Therefore, in this study, we analyzed the EC and IAEA methodology for derivation of weighting factor and used the latest nuclide data from ICRP 107 to derive weighting factors for dose coefficient. Weighting factor calculation is carried out through 1) selection of nuclide, 2) setting of evaluation period, and 3) derivation based on ICRP 107 radionuclide data. Firstly, in order to derive the weighting factor, we need to select the radionuclides whose dose contribution should be considered. If the half-life of progeny radionuclides sufficiently short compared to the parent radionuclide to achieve radioactive equilibrium, or if the dose coefficient is greater of similar to that of the parent radionuclide and cannot be ignored, the dose contribution of the progeny radionuclide should be considered. In order not to underestimate the dose contribution of progeny radionuclides, the weighting factors for the progeny nuclides are taken as the maximum activity ratio that the respective progeny radionuclides will reach during a time span of 100 years. Finally, the weighting factor can be derived by considering the radioactivity ratio and branch fraction. In order to calculate the weighting factor, decay data such as the half-life of the radionuclide, decay chain, and branch fraction are required. In this study, radionuclide data from ICRP 107 was used. As a result of the evaluation, for most radionuclides, the weighting factors were derived similarly to the existing EC and IAEA weighting factors. However, for some nuclides, the weighting factors were significantly different from EC and IAEA. This is judged to be a difference in the half-life and branch fraction of the radionuclide. For example, in the case of ^{95}Zr , the weighting factor for ^{95}mNb showed a 35.8% difference between this study and previous study. For ICRP 38, when ^{95}Zr decays, the branch fraction for $^{95\text{m}}\text{Nb}$ is 6.98×10^{-3} . In contrast, for ICRP 107, the branch fraction is 1.08×10^{-2} , a difference of 54.7%. Therefore, the weighting factor for the dose coefficient based on ICRP 107 data may differ from existing studies depending on the half-life and decay information of the nuclide. This suggests the need for a weighting factor based on the latest nuclide data. The results of this study can be used as a basis for the consideration of dose contributions for progeny radionuclides in various dose assessments.

Acknowledgements

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Keywords: Dose coefficient, Weighting factor, Dose assessment, Progeny radionuclide, Parent radionuclide

Development of a LabVIEW-Based Virtual Instrument System for the Automated Separation of Cesium Isotopes in Seawater

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The radioactive cesium, released from the normal operation or the accidental operation of nuclear facilities, should be regularly monitored for environmental regulatory compliance. The $^{135}\text{Cs}/^{137}\text{Cs}$ isotopic ratios, potentially useful for long-term tracking Cs transport in seawater, can be used as a tool of understanding how radionuclides are transported from different nuclear production source terms and distributed in the ocean.

The ultra-high sensitive mass spectrometers (TIMS, SF-ICP-MS and TQ-ICP-MS) have been used to measure the $^{135}\text{Cs}/^{137}\text{Cs}$ isotopic ratios. However, the radiochemical separation of Cs from the seawater matrix is essential for the analysis of Cs using the mass spectrometers.

An automated radiochemical procedure for the separation of Cs in seawater was developed for the analysis of $^{135}\text{Cs}/^{137}\text{Cs}$ isotopic ratios using a sequential column chromatography with AMP-PAN and AG50Wx8 cation exchange resins.

National Instrument's LabVIEW is a graphical programming language and a powerful tool for the instrument control. A virtual instrument system for the automated separation of cesium isotopes was developed by the state machine of the fundamental design patterns in LabVIEW.

In this study, the conceptual designs of an automated separation system of cesium isotopes, its virtual instrument system based on the LabVIEW state machine architectures and an automated radiochemical procedure were described for the purification of cesium isotopes at trace levels found in seawater discharged from the various nuclear facilities.

Acknowledgements

This work was supported by the research program of the Korea Atomic Energy Research Institute.

Keywords: LabVIEW, Virtual instrument, Automated separation, $^{135}\text{Cs}/^{137}\text{Cs}$ isotopic ratio, Seawater

A Study on Emergency Plans During the Decommissioning of Nuclear Power Plants

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South Korea's first commercial nuclear reactor, Kori Unit 1, was permanently shut down in 2017, and preparations are currently underway for its decommissioning. After the permanent shutdown, the spent nuclear fuel from the reactor core is removed and stored in a spent fuel storage facility. Subsequently, steps are taken for its permanent disposal, and if a permanent disposal site is not determined, it is stored in an interim storage facility (or temporary storage facility). Therefore, the activation criteria for radiation emergency plans vary depending on the movement of spent nuclear fuel and the storage location.

In this study, it reviewed emergency plans in the U.S. NRC Regulatory Guide (Draft) titled 'Emergency Planning for Decommissioning Nuclear Power Reactors' to determine the requirements for radiation emergency plans needed for decommissioned nuclear power plants. Additionally, by examining emergency plans applied to decommissioning nuclear power plants in the United States, this study identified emergency plan requirement that could be applicable to future decommissioned nuclear power plants in South Korea.

This study will contribute to the establishment of appropriate radiation emergency plans for decommissioning nuclear power plants in Korea for providing accurate information on overseas cases and relevant guidelines.

Acknowledgements

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Keywords: Decommissioning nuclear power plants, Emergency plan, Radiological emergency plan

Development of Wind Rose for Enhancing Public Protective Measures

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When occurring at a nuclear power plant (NPP) by accidents, accurate prediction and identification of the process of radioactive material dispersing into atmosphere is important to protect public and environment. Atmosphere dispersion of radioactive materials is significantly influenced by wind direction and wind speed. The government and nuclear operator continuously monitor wind data at nuclear sites through meteorological tower to prepare for such accidents involving the release of radioactive materials.

The purpose of this study is to construct wind rose diagrams at 5 NPP sites (Kori, Saewool, Wolsong, Hanbit, Hanul). Wind roses serve as invaluable tool for identifying wind patterns in each region and visualizing wind directions. This can be utilized to predict the dispersion pathway and extent range of radioactive materials carried by the wind. This program will take on the role of establishing appropriate evacuation routes or shelter locations for residents when reliable wind data is not immediately available during an NPP accident.

The wind data used in the study was collected from a meteorological tower located at the NPP site, and measurements were taken at 1-hour intervals for each operation over a period of ten years. The collected data underwent preprocessing, followed by the development of Python code to render the wind rose diagrams in an interpretable format.

The future direction of this study will be focused on enhancing this program by integrating geographical mapping capabilities. With these advancements, it will become feasible to superimpose shelter positions on a map in accordance with prevailing wind directions. These improvements will contribute to the development of additional protective measures for residents and the proposal of alternative shelter options in response to potential radioactive material releases.

Acknowledgements

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Keywords: Prediction, Wind rose, Dispersion patterns, Radiological emergency, Python

Consideration on Scope of Radiation Protection Plan for Decommissioning Project

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According to IAEA GSR Part.6, Decommissioning is carried out on the basis of planning and evaluation to ensure safety, protection of workers, public, and environment. Then, the decommissioning project of nuclear facility includes a radiation protection plan that reflects the regulatory requirements and international recommendations of each country and the internal regulations of the licensee. The scope of the radiation protection plan covers all radiation activities related to the dismantling and disposal of contaminated facilities subject to decommissioning.

Radiation protection applications in the United States, a country with previous experience in decommissioning nuclear facilities, include 10 CFR 20 for NRC management facilities and 10 CFR 835 for facilities under DOE. In this study, we analyzed two cases of decommissioning plans to which NRC regulations are applied.

In 1992, Yankee Atomic Electric Company (YAEC), the licensee of Yankee Nuclear Power Station (YNPS), notified NRC of the permanent shutdown of YNPS and submitted decommissioning plan accordingly. This decommissioning plan consists of a total of 9 chapters, and section 3.2 describes the radiation protection of decommissioning workers. The contents of the radiation protection program consist of 16 subsections.

Another case is the decommissioning work plan of U.S. Navy Surface Ship Support Barge (SSSB), which used in Virginia to support the refueling of the U.S. Navy's reactor vessel. This document was developed based on the NUREG-1757 and was revised in 2021 after receiving NRC comment. SSSB's project radiation protection plan is described in appendix 1, and the contents consist of a total of 28 sections except for reference.

In Korea, decommissioning plan is developed in accordance with "Standard Format and Content of the Decommissioning Plan for Nuclear Facilities". According to this regulation, the radiation protection plan for licensing documents submitted at the time of application for approval of decommissioning execution shall describe the organization and functions for implementing of plan, methods, cycles and procedures for performing radiation protection and radiological monitoring. Also, the safety review guidelines of regulatory body also require radiation protection plans and procedures to ensure ALARA activities during decommissioning.

In the case of the final decommissioning plan of Kori-1, which is currently submitted to regulatory body for licensing review, the decommissioning radiation protection plan is divided into 8 sections.

Although the classification criteria for the radiation protection plan categories described above facilities are different, it could be seen that the following 7 contents are included in common: (a) ALARA application and organization for implementation, (b) Management of radiation control area, (c) Process of radiation work, (d) Radiation and contamination control, (e) Personnel radiation exposure monitoring, (f) Radioactive material management, (g) Radiation protection training.

Keywords: Decommissioning, Radiation protection plan, Project plan

Development of a New Annual Dose Evaluation System Using On-Site Network Data

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A new annual dose evaluation system called E-DOSE has been developed. The system is based on the methodology of the previous version, K-DOSE60, which uses the dose evaluation methods of the International Commission on Radiological Protection (ICRP-60). However, E-DOSE is coded in ABAP to be compatible with the KHNP's enterprise resource planning (ERP) system, SAP. This allows E-DOSE to use the real-time data from SAP, which minimizes the need for user intervention. The socio-environmental data, which was previously managed by the staff of each plant sites, can now managed in the system in a centralized manner. This is a significant improvement over the previous system, as it reduces the risk of errors and makes it easier to track and manage data. The system also automatically generates the reports required by regulations. E-DOSE is expected to minimize the occurrence of human errors in preparing and managing the input data. This is because the system uses the data from SAP, which is less prone to errors than manually entered data. Additionally, the automatic generation of reports reduces the risk of errors in report preparation. E-DOSE is also expected to improve work efficiency. This is because the system automates many of the tasks involved in annual dose evaluation, such as data entry, calculation, and report generation. Overall, E-DOSE is a significant improvement over the previous annual dose evaluation system. It is more efficient, accurate, and user-friendly.

Keywords: Dose evaluation system, Socio-environmental data, ICRP-60, SAP

Medical Response to the Injured in a Radiological Emergency at the Korea Atomic Energy Research Institute

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The nuclear licensee must ensure that the nuclear or radiological emergency preparedness and response organization is explicitly defined and staffed with adequate numbers of competent and assessed personnel for their roles. This paper describes the responsibilities of medical and support personnel for the medical action of casualties in the event of a radiological emergency at the KAERI.

Currently, there is one medical personnel (nurse) in KAERI, and a total of eight medical support personnel are designated for medical response in the event of a radiological emergency. These medical support personnel are designated as one or two of the on-site response personnel at each nuclear facility, operating as a dedicated team of A, B (4 people each). In the event of a radiological emergency, not all medical support personnel are mobilized, but members of the dedicated medical team, which includes the medical support personnel of the nuclear facility where the accident has occurred, are summoned. Medical and support personnel will first gather in the onsite operational support center (OSC)/technical support center (TSC) to prepare and stand by for the medical response to injured when a radiological emergency is declared. They should take radiation protective measures, such as wearing radiation protective clothing and dosimeters, before entering the onsite of a radiological emergency, because injuries sustained during a radiological emergency may be associated with radioactive contamination. In the event of an injury, direct medical treatment such as checking the patient's vitals, first aid, and decontamination will be carried out by medical personnel, while support personnel are mainly responsible for contacting the transfer hospital, reporting the patient's condition, accompanying the ambulance, filling out the emergency medical treatment record, and supporting medical personnel. In order to respond appropriately to the occurrence of injuries, we regularly conduct emergency medical supplies education and medical training for medical support personnel to strengthen their capabilities.

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Keywords: Nuclear emergency preparedness, Radiological emergency, Injured, Medical response, Medical support personnel

Improvements to KAERIX for Effective Information Exchange in Radiological Emergencies

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In the event of a radiological emergency at a nuclear facility, the exchange of information on the accident situation is very important in the response process. For this reason, international organizations such as the IAEA and the EU operate systems to exchange information in the event of a radiological emergency. In south korea, the emergency response information exchange system (ERIX) developed by KINS is operated for use by the national radiological emergency response organization. The ERIX enables the exchange of emergency response information between organizations such as the government, nuclear operators, local authorities, KINS and KIRAMS. The KAERI has developed the KAERI emergency response information exchange system (KAERIX) for the exchange of accident information and emergency response information between the emergency response organizations of the KAERI in the event of a radiological emergency. This system is web-based using HTML, runs on internal network and is only available to KAERI staff.

Recently, as the need for optimizing and upgrading KAERIX has arisen, improvements have been derived. The main improvement is optimizing KAERIX for Microsoft Edge to minimize errors. At present, it is optimized for Internet Explorer, but optimizing it for Microsoft Edge mode has become essential due to Microsoft discontinuing support for Internet Explorer. Another major improvement involves adding functions in ERIX to KAERIX, such as displaying the deletion/correction status of input information and providing notifications for important information registration. Ultimately, KAERIX will be upgraded and optimized in 2024, reflecting improvements.

Acknowledgements

This study has been conducted as a part of the project of emergency preparedness for nuclear research facilities in Korea Atomic Energy Research Institute (No. 521530-23).

Keywords: KAERIX, ERIX, Radiological emergency, Nuclear emergency preparedness

Study on Optimization Gaseous Effluent Evaluation Point of Radiological Environmental Impact Assessment

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Domestic nuclear power plants conduct radiological environmental impact assessments every year in accordance with the Nuclear Safety and Security Commission (NSSC) notice. Among them, gaseous effluents are evaluated for their effects due to inhalation, external exposure in the air, exposure from ground surface deposits, food intake. In order to evaluate the impact of this exposure pathway, an evaluation point for each pathway must be selected. In the case of evaluation points, each country has different evaluation points. In the case of Korea, the evaluation point is calculated on the assumption that one lives 365 days a year at the EAB and consumes food from the nearest production area. In the case of the United States, external exposure and inhalation are evaluated at the site boundary or the nearest residential area, and food intake is evaluated by assuming that food produced in the nearest residential area or the nearest production area is consumed.

Currently, the dose evaluation is optimized and selected so that EAB evaluation point for each site includes 16 direction evaluation points for each unit. In the E-DOSE60 program currently under development, the evaluation point was selected by calculating 16 direction x number of units without optimization. The food intake evaluation point was selected as the point that satisfies the minimum farmland area of the U.S. NRC NUREG-1301 and is the shortest distance from the site. The location of the production point from multiple units is included all 16 directions for each unit and quantity of evaluation points was optimized to satisfy the shortest distance.

It can contribute to improving the reliability of the E-DOSE60 program currently under development by selecting new evaluation points for evaluating inhalation and external exposure evaluation and selecting optimized dose evaluation points for each site for evaluation by ingestion.

Keywords: Gaseous effluent, ODCM, Exposure pathway, E-DOSE60

Applicability Analysis of Acoustic Emission and Ultrasonic Techniques for Leak Detection in ACPF Mock-up Facility

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Korea Atomic Energy Research Institute (“KAERI”) has been developing various studies related to the nuclear fuel cycle. Among them, KAERI was focusing on the pyroprocess, which recycles some useful elements while reducing the volume and toxicity of spent nuclear fuel (SNF). Pyroprocess involves the handling of SNF, which cannot be handled directly by the facility worker. Therefore, SNF is handled and processed through remote handling device within a shielded facility such as a hot cell. Nuclear Facilities with such hot cells are called nuclear fuel cycle facilities, and unlike other facilities, heating, ventilating, and air conditioning (HVAC) system are particularly important in nuclear fuel cycle facilities to maintain the atmosphere in the hot cell and remove radioactive materials. In addition, due to the nature of the pyroprocess, which uses molten salt, corrosion is a problem in air atmosphere, so the process can only be carried out in an inert gas atmosphere.

KAERI has a nuclear fuel cycle facility called the Irradiation Material Examination Facility (IMEF), and has built and operated the ACPF inside the IMEF, which operates an inert atmosphere hot cell for the demonstration of the pyroprocess. For efficient process development of the pyroprocess, it is necessary to put the developed equipment into the hot cell, which is a radiation-controlled area, after sufficient verification in a mock-up facility. For this purpose, the ACPF mock-up facility, which simulates the system, space, and remote handling equipment of the ACPF, is operated separately in the general laboratory area. The inert gas conditioning system of the ACPF consists of very complex piping, blowers, and valves, requires special attention to maintenance. In addition, if there is a small leak in the piping within these valves or piping, radioactive materials can be directly exposed to facility workers, so continuous monitoring and maintenance are required to prevent accident.

In this study, the applicability of acoustic emission technology and ultrasonic technology for leak detection in the inert gas conditioning system of ACPF mock-up facility was investigated. For this purpose, new bypass pipes and valves were installed in the existing system to simulate the leakage of pipes and valves. Acoustic emission sensors are attached directly to pipes or valves to detect signals, while ultrasonic sensors are installed at a distance to detect signals. The optimal parameters of each technology to effectively suppress background noise were derived and, and the feasibility of identifying normal and abnormal scenarios in the system was analyzed.

Acknowledgements

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Keywords: Acoustic emission, Ultrasonic, Pyroprocess, ACPF, Leak detection

Radiation Hardness Requirements Analysis for Pyroprocess Facility

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Korea Atomic Energy Research Institute (“KAERI”) has been developing pyroprocess technology for the sustainable use of nuclear energy and radioactive waste reduction, and is conducting design studies for a Pyroprocess Commercializing Research Facility (PCRF).

High-level radioactive materials such as spent nuclear fuel, which are handled in the hot cell of the PCRF, physically change materials directly or cause chemical changes through ionization or excitation depending on the energy and types of radiation. Therefore, all facilities, including process equipment and remote handling equipment, installed into the hot cell must be evaluated for radiation hardness to be maintained in the radiological environment so that processes can proceed throughout the design life of the facility. In addition, as the maintenance paradigm has recently shifted from corrective maintenance to predictive maintenance, it is necessary to know in advance the condition of the equipment or facility in the radiological environment.

In this study, an analysis of the radiation environment of the hot cell in the PCRF was conducted through source term, and the radiological dose impact was evaluated through the results of irradiation experiments of major components by reference data. Then, the actual dose contribution was identified through dose assessment using the MCNP code based on the pyroprocess equipment, and the radiation hardness requirements for the facility and equipment in the hot cell were derived by the above results.

Acknowledgements

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Keywords: Radiation hardness, Hot cell, Pyroprocess, Design requirements

A Case Study on Systems for Measurement and Classification of Contaminated Soil in Real-Time

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When decommissioning a nuclear power plant, it is expected that clearance or radioactive waste (e.g., soil, concrete, metal, etc.) below the low-level will be generated in a short period on a large scale. Among the various types of waste, most of the contaminated soil is known to be classified as clearance or the (very) low-level radioactive waste. Accordingly, an accurate measurement and classification of contaminated soil in real-time during the decommissioning process can efficiently reduce the amount of soil waste and the possibility of contamination diffusion. However, in order to apply a system that measures and classifies contaminated soil in real-time according to the level of contamination to the decommissioning site, a demonstration is required to evaluate whether the system is applicable to the site. In this study, to establish requirements for determining the applicability of the system to the decommissioning site, preceding cases from countries with abundant decommissioning experience were investigated. For example, MACTEC of the U.S. demonstrated the developed system at the Saxton nuclear power plant in the U.S. and confirmed that the amount of soil that can be analyzed per hour in the system is affected by radionuclides, minimum detectable activity (MDA), and applicable volume. In the future, therefore, we will utilize the result of this study to develop the requirements of demonstrating the system for measurement and classification of contaminated soil in real-time.

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Keywords: Decommissioning, Contaminated soil, Radioactive waste, Clearance, Soil measurement & classification system

Review of Environmental Monitoring Activities During the Nuclear Emergency

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The primary objective of radiological environmental monitoring after a radiological emergency at a nuclear facility is acquisition of background data for the determination of protective actions for the population and the comprehensive assessment of the impact on the population residing in proximity to the nuclear facility. The responsible entities engaged in the conduct of the radiological environmental monitoring encompass government organization and nuclear licensees, operating in strict adherence to the national radiological disaster prevention framework.

In accordance with the national radiological disaster prevention framework, radiation environmental monitoring is executed through the deployment of emergency response organization, and recurrent exercise drills aimed at augmenting responsible capabilities. In the context of radiation environmental monitoring, it is necessary to specify measurement parameters, monitoring location, and methodological protocols for each stage, considering potential exposure pathways. In terms of equipment, it is important to utilize mobile assets such as aerial or vehicle surveys for rapid and accurate radiation environment monitoring. Radiation disaster drills are regularly conducted, and the radiation environment monitoring field is also regularly trained to enhance response capabilities. The scale of these drills may vary, ranging from exclusive participation by nuclear licensees to joint exercises conducted by governmental agencies. This iterative process of periodic drills and equipment enhancements has led to a progressive augmentation of environmental monitoring capabilities, ensuring a well-coordinated orchestration of radiation monitoring within the framework of radiation protection.

Notwithstanding these achievements, challenges in public communication regarding the decision to take protective actions and the dissemination of information to the public. Considering that the purpose of radiation environmental monitoring extends beyond safeguarding public health; it also serves to alleviate public anxiety. In the future, public communication between these stakeholders should also be included in disaster drill programs to ensure proper consultation between each stakeholder during drills and to build understanding and trust in radiation environmental monitoring.

This is expected to improve the quality of radiation environmental monitoring response capabilities.

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, Exercise drill

Radiological Dose Assessment Resulting From Routine and Accidental Release of Radioactive Waste Management Facilities in KAERI

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The Korea Atomic Energy Research Institute (KAERI) has facilities that are operated for the purpose of treating radioactive wastes and storing drums before sending them to a disposal site. Domestic regulations related to nuclear facility require radiological dose assessment resulting from release of gaseous radioactive effluent of nuclear facilities. In this study, ICRP-60-based dose conversion factors were applied to evaluate the radiation dose to residents in the event of operation and accident for the radioactive waste management facilities in KAERI. The radioactive gaseous effluent generated from each facility diffuse outside the exclusion area boundary (EAB), causing radiation exposure to residents. To evaluate the external exposure dose, the exposure pathways of cloudshine and radioactive contaminated soil were analyzed. The internal exposure dose was estimated by considering the exposure from respiration and ingestion of agricultural and livestock products. The maximum individual exposure dose was evaluated to be 1.71% compared to the dose limit. The assumed situation used for accidental scenarios are as follows; A fire inside the facility and falling of radioactive waste drum. It was a fire accident that caused the maximum exposure dose to individual and population living within an 80 km radius of the site. At the outer boundary of the low population zone (LPZ), the maximum effective dose and thyroid equivalent dose were estimated as 8.92×10^{-6} % and 5.29×10^{-6} %, respectively, compared to the dose limit. As a result of evaluating the radiological exposure dose from gaseous emissions, the radioactive waste treatment facilities and its supplementary facilities meet the regulations related to nuclear facility, and are operated safely in terms of radiological environmental impact assessment.

Keywords: Dose assessment, Residential exposure, Radioactive waste management facility

A Study on the CZT Detector Calibration Method Through HPGe Detector Calibration Method

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The dismantling nuclear power plant is expected to continue to change the radiation working environment compared to the operating nuclear power plant. Contamination monitors and survey meters currently in use have limitations in accurate analysis source term and dose rates for continuous changes in radiation fields at dismantling sites. Due to these limitations, the use of semiconductor detectors such as HPGe and CZT detectors with excellent energy resolution and portability is increasing. The CZT detector performs as well as the HPGe detector, but there is no proven calibration procedure yet. Therefore, in this study, the HPGe calibration method was reviewed to derive implications for the CZT detector calibration method.

The operating principle of a semiconductor detector that measures gamma emission energy converts them into electrical signals is the same. Two calibrations of HPGe detectors are performed according to the standard calibration procedure for semiconductor detectors for gamma-ray measurement issued by the Korea Association of Standards & Testing Organizations. The first is an energy calibration that calculates gamma-ray peak position measurements and relational expressions using standard source term that emit gamma-rays. The channel values for energy are measured using certified reference source term to determine radionuclides by identifying channels corresponding to the measured peak energy values. The second is the measurement efficiency of measuring the coefficient calibration device, which measures gamma rays emitted from the standard source term. The detector efficiency by sample or distance is measured in consideration of the shape, size, volume, and density of the calibration device.

The HPGe detector performs calibration once every six months through a verified calibration method and is being used as a source term analyzer at the power plant. The CZT detector may also establish a procedure for identifying peak positions through energy calibration and calculating radioactivity through efficiency calibration. This will be a way to expand the usability of semiconductor detectors and further monitor radiation in a more effective way.

Keywords: Calibration method, CZT, HPGe, Nuclear power plant decommissioning

Assessment of Worker Radiation Exposure for the Cutting of Steam Generator Tubes

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The first commercial operation of Kori-1, which commenced in April 1978, was permanently shut down in June 2017, with plans for immediate dismantling. The decommissioning process of nuclear power plants generates a substantial amount of radioactive waste and poses significant radiation exposure risks to workers. Radioactivity is widely distributed throughout the primary coolant system of the reactor, including the reactor pressure vessel (RPV), steam generator (SG), and pressurizer. In particular, the SG has a considerable size and complex geometry, weighing approximately 326 tons and having a volume of 400 m³. The SG tubes are known to contain high levels of radioactivity, leading to significant radiation exposure to workers during the dismantling process. Therefore, this study aims to evaluate the workers' radiation exposure during the cutting of SG tubes, which account for approximately 95% of the total radiation dose in the SG.

Firstly, the CRUDTRAN code, developed to predict the behavior of soluble and particulate corrosion products in a PWR primary coolant system, is used to estimate the radioactive inventory in the SG tubes. Based on decontamination factors (DF) obtained in the SG tubes through overseas experience, the expected reduction in radioactivity during the Kori-1 reactor's full-system decontamination (FSD) process is considered in the CRUDTRAN results. These results are then processed to estimate the radioactivity in both the straight and bent sections of the tubes. Subsequently, these radioactivity values are used as inputs for the MicroShield code to calculate the worker radiation exposure during the cutting of both straight and bent sections of the tubes. The cutting process assumes that each SG tube section is cut in a separate, shielded area, and the radiation exposure is assessed, taking into account factors such as cutting equipment, cutting length, working hours, and working distance.

This study evaluates the worker radiation exposure during the cutting of SG tubes, which are expected to have a significantly high radioactivity due to chalk river unidentified deposit (CRUD). This assessment also considers the reduction in radioactivity within the steam generator tubes resulting from the FSD process. Consequently, it enables an examination of how worker radiation exposure varies based on the extent of FSD. This study may provide valuable insights for determining the scope and extent of the FSD process and the development of shielding methods during the dismantling of SG tubes in the future.

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Keywords: Kori-1, Steam generator (SG), SG tube, Worker radiation exposure, Full-system decontamination (FSD), CRUDTRAN, MicroShield

Diagnosis of a HVAC System via Prognostics and Health Management (PHM) for Soundness and Safety of Fuel Cycle Facilities

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One of the important components of a nuclear fuel cycle facility is a hot cell. Hot cells are engineered robust structures and barriers, which are used to handle radioactive materials and to keep workers, public, and the environment safe from radioactive materials. To provide a confinement function for these hot cells, it is necessary to maintain the soundness of the physical structure, but also to maintain the negative pressure inside the hot cell using the operation of the heating, ventilation, and air conditioning (HVAC) systems. The negative pressure inside the hot cells allows air to enter from outside hot cells and limits the leakage of any contaminant or radioactive material within the hot cell to the outside. Thus, the HVAC system is one of the major components for maintaining this negative pressure in the hot cell. However, as the facility ages, all the components of the hot cell HVAC system are also subject to age-related deterioration, which can cause an unexpected failure of some parts. The abnormal operating condition from the failure results in the increase of facility downtime and the decrease in operating efficiency. Although some major parts are considered and constructed in redundancy and diversity aspects, an unexpected failure and abnormal operating condition could result in reduction of public acceptance and reliability to the facility. With the advent of the 4th Industrial Revolution, prognostics and health management (PHM) technology is advancing at a rapid pace. Korea Hydro & Nuclear Power, Siemens, and other companies have already developed technologies to constantly monitor the integrity of power plants and are applying the technology in the form of digital twins for efficiency and safety of their facility operation. The main point of PHM, based on this study, is to monitor changes and variations of soundness and safety of the operation and equipment to analyze current conditions and to ultimately predict the precursors of unexpected failures in advance. Through PHM, it would be possible to establish a maintenance plan before the failure occurs and to perform predictive maintenance rather than corrective maintenance after failures of any component. Therefore, it is of importance to select appropriate diagnostic techniques to monitor and to diagnose the condition of major components using the constant examination and investigation of the PHM technology. In this study, diagnostic techniques are investigated for monitoring of HVAC and discussed for application of PHM into nuclear fuel cycle facilities with hot cells.

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Keywords: Prognostics and health management (PHM), Fuel cycle facility, Hot cell, HVAC system, Constant monitoring, Safety

MCNP Simulation Based Optimization of Prussian Blue for Cesium Removal

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The nuclear fuel that melted during the Fukushima nuclear accident in 2011 is still being cooled by water. In this process, contaminated water containing radioactive substances such as cesium and strontium is generated. The total amount of radioactive pollutants released by the natural environment due to the nuclear accident in Fukushima in 2011 is estimated to be 900 PBq, of which 10 to 37 PBq for cesium. Radioactive cesium (^{137}Cs) is a potassium analog that exists in the water in the form of cations with similar daytime behavior and a small hydration radius and is recognized as a radioactive nuclide that has the greatest impact on the environment due to its long half-life (about 30 years), high solubility and diffusion coefficient, and gamma-ray emission.

In this study, alginate beads were designed using Prussian blue, known as a material that selectively adsorbs cesium for removal and detection of cesium. To confirm the adsorption performance of the produced Prussian blue, immersion experiments were conducted using Cs standard solution, and MCNP simulations were performed by modeling 1L reservoir to conduct experiments using radioactive Cs in the future. An adsorption experiment was conducted with water containing standard cesium solution using alginate beads impregnated with Prussian blue. The adsorption experiment tested how much cesium of the same concentration was adsorbed over time. As a result, it was found that Prussian blue beads removed about 80% of cesium within 10-15 minutes. In addition, MCNP simulation was performed using a 1 L reservoir and a 3inch NaI detector to optimize the amount of Prussian blue. The results of comparing the efficiency according to the Prussian volume was shown. It showed that our designed system holds great promise for the clean-up and detection of radioactive cesium contaminated seawater around nuclear plants and/or after nuclear accidents. Thus, this work is expected to provide insights into the fundamental MCNP simulation based optimization of Prussian blue for cesium removal and this work based MCNP simulation will pave the way for various practical applications.

Keywords: MCNP, Simulation, Optimization cesium, Prussian blue, Adsorption

Radiological Characteristics of the Havar Foil in the Cyclotron Target

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The cyclotron is an apparatus used for the production of radioactive isotopes through nuclear reactions, resulting in the inevitable emission of neutrons. Consequently, surrounding components become activated. The purpose of this study was to investigate the radiological characteristics of Havar foil, a periodic replacement part of the Targetry system. In this study, radioactivity and radiation dose were calculated based on the time of Havar foil replacement and equipment dismantling. The time to dismantle the equipment was set at one year after the equipment was shut down, based on the recently used 1g of Havar foil. All simulations were performed using the FLUKA program. First, in the simulation results, 11 elements (Re, W, Tc, Nb, Cu, Ni, Co, Fe, Mn, Cr, V) were converted into 36 radioisotopes by activation based on the replacement period. Based on radioactivity levels, major isotopes included ⁵²Mn (77.63%), ⁵⁶Co (13.36%), ⁹⁶Tc (2.4%), and ⁹⁵Tc (1.80%). Based on radiation dose rates, ⁵²Mn (82.66%) and ⁵⁶Co (13.45%) exhibited the highest levels. Furthermore, the dose rates at distances of 10 cm, 50 cm, and 100 cm were found to be 1.36E+1 mSv/hr, 2.24E+00 mSv/hr, and 8.80E-01 mSv/hr, respectively. Second, as of the time the equipment was dismantled, 20 radioactive isotopes of 10 elements, excluding short-lived nuclides, were generated. In terms of radioactivity, ⁵⁶Co (45.83%), ⁵⁵Fe (19.73%), ⁵⁷Co (14.48%), and ⁵⁴Mn (13.50%) were prominent. Regarding radiation dose rates, ⁵⁶Co (92%) and ⁵⁴Mn (7.32%) exhibited higher levels. Dose rates at distances of 10 cm, 50 cm, and 100 cm were calculated at 5.31E-01 mSv/hr, 8.80E-02 mSv/hr, and 3.47 E-02 mSv/hr, respectively. Third, according to the radioactive waste classification standards in the replacement and decommissioning stages, Havar foil was predicted to be low-level radioactive waste in terms of radioactivity. In addition, it was derived that a cooling period of approximately 12 years is necessary to satisfy the allowable dose for clearance level waste. In conclusion, Havar foil, which is periodically generated as radioactive waste, can cause radiation exposure to replacement workers. Therefore, special and careful management is required for the Havar foil of the cyclotron.

Keywords: Cyclotron, Havar foil, Activation, Radiological characteristics

Safety Perceptions and Training Experience of Emergency Personnel Regarding the Emergency Preparedness for Nuclear Facilities

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The objective of this study is to investigate the safety awareness and effectiveness of the education and training for employees engaged in radiological emergency organization of the Korea Atomic Energy Research Institute (KAERI). In 2022, the questionnaire for the education satisfaction survey was revised to regularly evaluate the effect of education on perceptions of importance on emergency preparedness for nuclear research facilities. In line with, a standard questionnaire was created which covers 3 factors and 9 attributes, and the evaluation indicator is based on a 5-point Likert scale. In 2023, the education on radiological emergency preparedness was conducted for 235 emergency staff. From May 24 to July 13, 2023, data was collected from a total of 235 emergency response personnels, including 28 new staffs and 207 maintenance staffs. As a result of response analysis, it was identified that education for radiological emergency response had a significant correlation with the promoting safety culture. It was found that senior emergency personnel with more years of experience are highly interested in radioactive disaster prevention and actively participate in and training. On the other hand, it was presented that new and less experienced groups tend to have a relatively high score of the risk perception of nuclear research facilities. Therefore, it is necessary to improve the practical curriculum in order to increase the participation of junior disaster prevention personnel in education and training, ensuring that they correctly recognize the risk of research facilities. This results are expected to be used to improve the quality of education and drills for radiological emergency response 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, Education for radiological emergency response, Safety perceptions

Analysis of the Off-site Consequence by the Occurrence Time of HFFF DBA

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To ensure the maintenance of the nuclear emergency response system, it is important to periodically conduct hazard assessments using up-to-date input variables. The results of this review are applied to drills and exercises, enabling the inspection of emergency plan and response procedures. Therefore, this study aims to analyze off-site consequences according to the occurrence time of the Design Basis Accident (DBA) for the Hanaro Fuel Fabrication Facility (HFFF) by using the recent site-specific meteorological data and to review the appropriateness of urgent protective measures. MELCOR and SafeHanaro computer codes were used for radiation source-term estimation and environmental impact assessment, respectively. It was assumed that radioactive materials are released into environment for 2 hours due to the fire during the nuclear fuel sieving process. The following 12 scenarios for each occurrence time period was selected (0 am, 2 am, 4 am, 6 am, 8 am, 10 am, 12 pm, 2 pm, 4 pm, 6 pm, 8 pm, 10 pm) and the effective dose and thyroid dose in early- and intermediate-phase were assessed. As a result, the most severe exposure-induced accident scenario is found to be as occurring at 0 am on July 15th, with the Most Exposed Individual (MEI) positioned 200 meters downwind from the facility. The committed effective dose for MEI is identified as to be $2.97\text{E-}02$ mSv which has a significant margin against the IAEA's (Generic Intervention Level) GIL and (Generic Criteria) GC. During the passage of the radio-active plume, the estimated effective dose and thyroid dose due to inhalation were $2.97\text{E-}02$ mSV (99.99%) and $5.06\text{E-}05$ mSv (99.77%), respectively. External exposure appeared to be negligible. Meanwhile, the thyroid dose is noticeably below the criteria for decision-making for distribution of Potassium Iodide (KI). Accordingly, in order for local residents to participate in the exercise and drills, it is essential to develop scenarios considering simultaneous emergencies at multi-facilities and late-night accidents. In conclusion, this results will be used to improve the exercise plans for enhancing the nuclear or radiological emergency competencies of the KAERI.

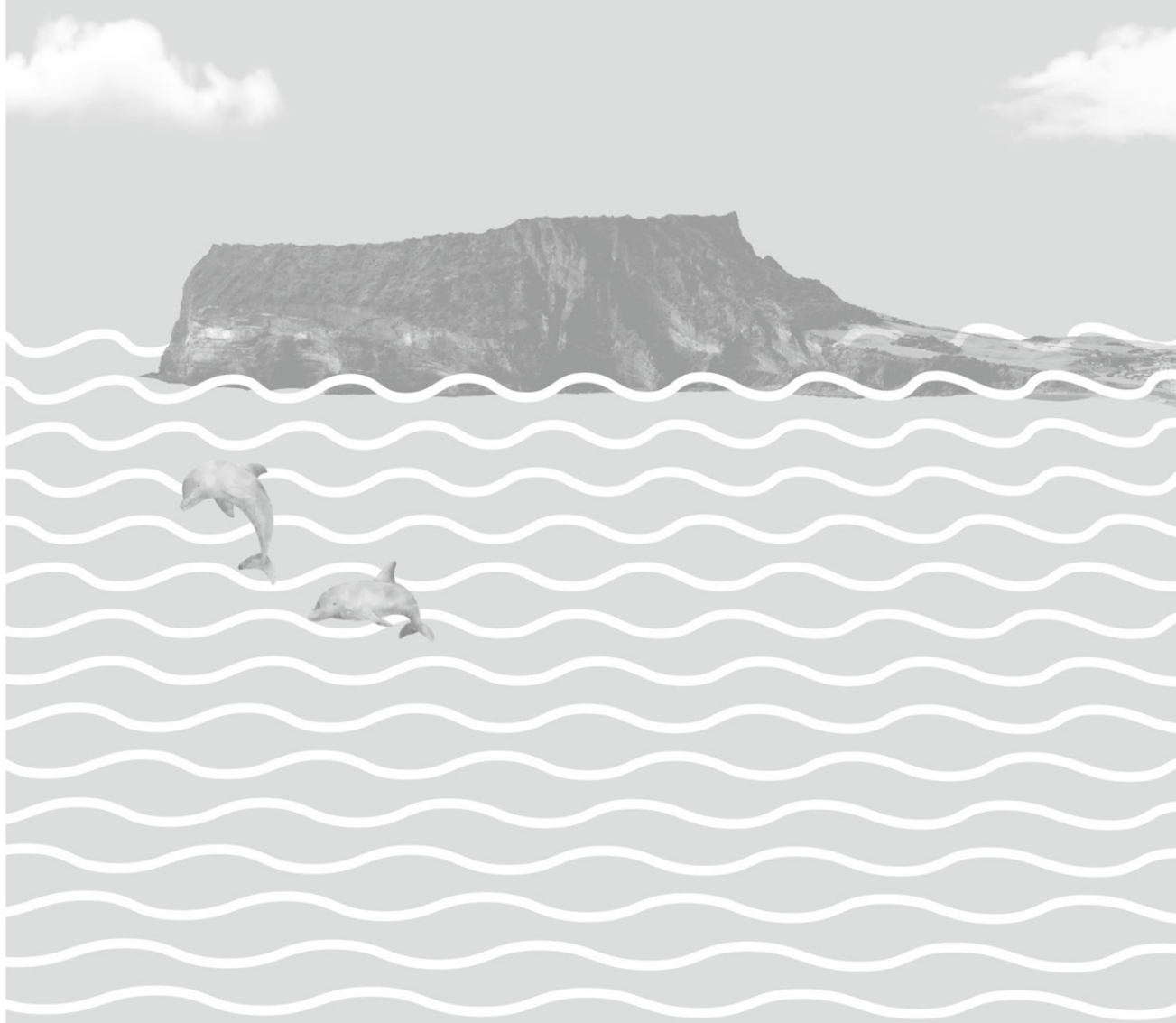
Acknowledgements

This study has been conducted as a part of the project of emergency preparedness for nuclear research facilities in KAERI.

Keywords: Fuel fabrication facility, Nuclear or radiological emergency, Emergency preparedness, Radiological protection, Emergency response exercise

7분과

방사화학 (ORAL)



Comparative Study for the Iodate Removal Mechanism Using the Layered Double Hydroxides Under Experimental and Computational Approaches

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Radioactive iodine released from nuclear power plants has been recognized to pose significant risks and environmental hazards. In response to these challenges, extensive investigations into iodine sorbents have been conducted with a particular focus on the utilization of layered double hydroxides (LDH) as a promising candidate. Herein, we have focused on the investigation of LDH materials featuring diverse transition metals for their synthesis, with specific emphasis on CoAl LDH for its proficiency in removing iodine species, particularly IO_3^- . Nevertheless, a comprehensive understanding of the removal mechanisms employed by these LDH materials remained elusive. Hence, the primary aim of this study is to elucidate the intricacies of the removal mechanisms through sorption tests, spectroscopic techniques, and theoretical chemistry analyses, subsequently contrasting the experimental outcomes with computational results.

For the experimental facet, the synthesis of CoAl LDH was conducted utilizing 0.15 mol L^{-1} of $\text{Co}(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ and 0.06 mol L^{-1} of $\text{Al}(\text{NO}_3)_3 \cdot 9\text{H}_2\text{O}$ to attain a molar ratio ($\text{M}^{2+}:\text{M}^{3+}$) of 2.5:1. Subsequently, pH-dependent IO_3^- sorption tests were carried out, coupled with X-ray absorption near-edge structure (XANES) and extended X-ray absorption fine structure (EXAFS) spectroscopy, facilitating the elucidation and discourse of the removal mechanism. The theoretical chemistry in this research harnessed ab initio molecular dynamics (AIMD) simulations for structural modeling, atomic density profiles, radial distribution function, analysis of oxide species, and MD-EXAFS spectrum analysis.

In summary, this study aims to elucidate iodine removal mechanisms using diverse experimental results, culminating in the revelation that ion-exchange with NO_3^- present in the interlayer predominates as the principal mechanism for IO_3^- removal. Notably, a distinct spectral feature at approximately 33,190 eV emerged, defying identification through XANES and EXAFS analyses conducted under experimental conditions. In the AIMD simulations, meticulous scrutiny of individual iodine atoms uncovered the prevalence of I–O and I–O–H molecular species, marked by interactions between O and H atoms, with a coordination number of $\text{I–O} \approx 3$. This transformation was primarily instigated by proton hopping. As a result, the comparative investigation reveals the dominance of IO_3^- intercalation in the CoAl LDH material with the potential to undergo a transformation to the I–O–H molecule upon interaction with protons.

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Keywords: Layered double hydroxides, Iodine, XAFS, AIMD simulations

A New Perspective- Analysis of Strontium-90

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The radioactive contamination in the ocean has raised significant concern on the environmental impact among Asian and Pacific countries since the Fukushima Daiichi Nuclear Power Plant accident (Mar 11, 2011). The first step in determining the contamination by the radioactive material is monitoring anomalies of environmental radioactivity of interest. As a result, each country has its own environmental radioactivity surveillance program.

Strontium-90 (half-life 28.8 y) is one of the radionuclides of high interest in the environment, owing to its high fission production rate and biological accumulation resulting from similar chemical behavior with calcium. The level of Strontium-90 in the seawater is very low, with a global average of about 1 mBq kg⁻¹. Consequently, it requires large volume of seawater sample, typically ranging from 40 L to 60 L. The purification of ⁹⁰Sr from seawater sample is challenging due to the high salinity and presence of stable Sr (about 7 ppm). Therefore, the conventional method for determining ⁹⁰Sr is time-consuming and labor-intensive work.

The author reported an advanced method, which is a more analyst-friendly and simpler method compared to the current method, for the determination of ⁹⁰Sr in seawater. This method focuses on the separation of ⁹⁰Y, which is equilibrium with ⁹⁰Sr, utilizing a commercialized extraction resin. As a result, it takes less than 3 hours to determine ⁹⁰Sr in 50 L of seawater sample and requires less labor. Additionally, this approach could be applied to the analysis of ⁹⁰Sr in radioactive waste

Acknowledgements

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Keywords: Strontium-90, Radioactivity, Secular equilibrium, Extraction chromatography

Improvement of ^{94}Nb Analytical Sensitivity in Radioactive Dry Active Waste With Sequential Chemical Separation

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To achieve permanent disposal of radioactive waste drums, the radionuclides analysis process is essential. A variety of waste types are generated through the operation of nuclear facilities, with dry active waste (DAW) being the most abundant. To perform radionuclides analysis, sample pre-treatment technology is required to transform solid samples into solutions. In this study, we developed a dry ashing-microwave digestion method and secured the reliability of the analysis results through a validity evaluation. Additionally, we conducted a comparative analysis of the radioactivity of ^{94}Nb nuclides with and without the chemical separation process, which reduced the minimum detectable activity (MDA) level by more than 65-fold for a certain sample.

Acknowledgements

This work was supported by the Nuclear Power Plant Decommissioning Technology Development Program (Project No. 77873-23) of the Korea Institute of Energy Technology Evaluation and Planning (KETEP). Sequential separation experiments were supported by the KAERI Institutional Program (Project No. 521320-23).

Keywords: Radiochemistry, Sequential chemical separation, Anion exchange resin, Gamma spectrometry, Minimum detectable activity

Exploring the Reaction Mechanism in Molten Salts Through Real-time Visual Monitoring System

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High-temperature molten salts not only demonstrate exceptional thermal and chemical stability but also offer significant advantages in catalyzing chemical reactions. Consequently, they have garnered attention as a promising medium for next-generation nuclear reactors and a wide range of electrochemical processes. Nevertheless, the challenging experimental conditions in molten salts make applying conventional analytical methods to understand reaction mechanisms a formidable task. This underscores the imperative need for more intuitive approaches to investigate molten salt chemistry. One of the simplest yet potent methods involves real-time visual monitoring of the reaction system as chemical reactions progress.

In light of this, we have developed an experimental system enabling real-time visual monitoring of the internal dynamics of molten salt media. This system can capture high-resolution videos and images within molten salts, surpassing existing methodologies. We have applied this system in various electrochemical experiments using the molten LiCl-KCl eutectic salt medium. Among them, this study primarily focuses on two challenging experimental scenarios that became comprehensible through our proposed system's application: (1) the transpassivation of Zr metal and the agglomeration of potassium hexachlorozirconate (K_2ZrCl_6) solid salt, and (2) the solvation of electrons during the oxidation of Li metal within the molten LiCl-KCl eutectic salt.

Acknowledgements

This work was supported by a grant from the National Research Foundation of Korea and Korean Ministry of Science and ICT [grant numbers 2017M2A8A5014801, 2021M2C7A1A0207 6339].

Keywords: Molten salt, Real-time monitoring, Zirconium, Transpassivation, Lithium, Solvation

Comparative Analysis of Alpha Activities of KAERI's Radioactive Waste Determined by Different Methods

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Alpha activities can be used for categorization, transportation, and disposal of radioactive waste generated from the operation of nuclear facilities including nuclear power plants. In order to transport and dispose of such low- and intermediate-level radioactive waste (LILW) to the Wolsong LILW Disposal Center (WLDC) at Gyeongju, the gross alpha concentration of an individual drum should be determined according to the acceptance criteria. In addition, when the gross alpha concentration exceeds 10 Bq/g, the inventory of the comprising alpha emitters in the waste is to be identified.

Gross alpha measurements using a proportional counter are usually straightforward, inexpensive, and high-throughput, so they are broadly used to assay the total alpha activity for environmental, health physics, and emergency-response assessments. However, several factors are thoughtfully considered to obtain a reliable approximate for the entire alpha emitters in a sample, which include the alpha particle energy of a particular radionuclide, the radionuclide that is used as a calibration standard, the uniformity of film in a planchet, time between sample collection and sample preparation, and time between sample preparation and counting.

Korea Atomic Energy Research Institute (KAERI) have evaluated the inventory of radionuclides in low-level radioactive waste drums to send every year hundreds of them to the WLDC. In this presentation, we revisit the gross alpha measurement results of the drums transported to WLDC in the past few years and compare them with the concentrations of alpha emitters measured from alpha spectrometry and gamma spectrometry. This study offers an insight into the gross alpha measurement for radioactive waste regarding calibration source, self-absorption effect, composition of alpha emitters, etc.

Acknowledgements

This work was supported by the KAERI Institutional Programs (Project Numbers: 521320-23 and 521330-23).

Keywords: Disposal, Repository, Low- and intermediate-level radioactive waste, Radionuclide inventory, Gross alpha, and gas-flow proportional counting

Recent Development of Measurement Methods of Difficult-to-Measure Nuclides in Radioactive Waste

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For the disposition of radioactive wastes generated from nuclear power plant, radioisotope inventory must be analyzed to determine an activity concentration of radionuclides. Radionuclides in low- and intermediate-low-level of radioactive wastes, however, can be easily classified to easy-to-measure (ETM) and difficult-to-measure (DTM) nuclides. ETM nuclides are gamma emitting nuclides that is relatively easy to measure because they do not need to be destroyed for the preprocessing. On the other hands, DTM nuclides are alpha and beta emitting nuclides that need to be destroyed for the preprocessing and also need chemical separation. Currently, measurement methods for DTM nuclides are developed and in this paper measurement methods of Fe-55, Ni-59, Ni-63, Sr-90 and Tc-99 will be introduced.

Keywords: Radioactive waste, Radiochemical analysis, Difficult-to-measure, Radioisotope inventory

~~Recovery Differences Based on the Mass of DMG and DIBK Resins in Column
Chromatography Separation Experiments of Fe⁵⁵ and Ni^{59/63}~~

~~Eun-Bin Mo~~

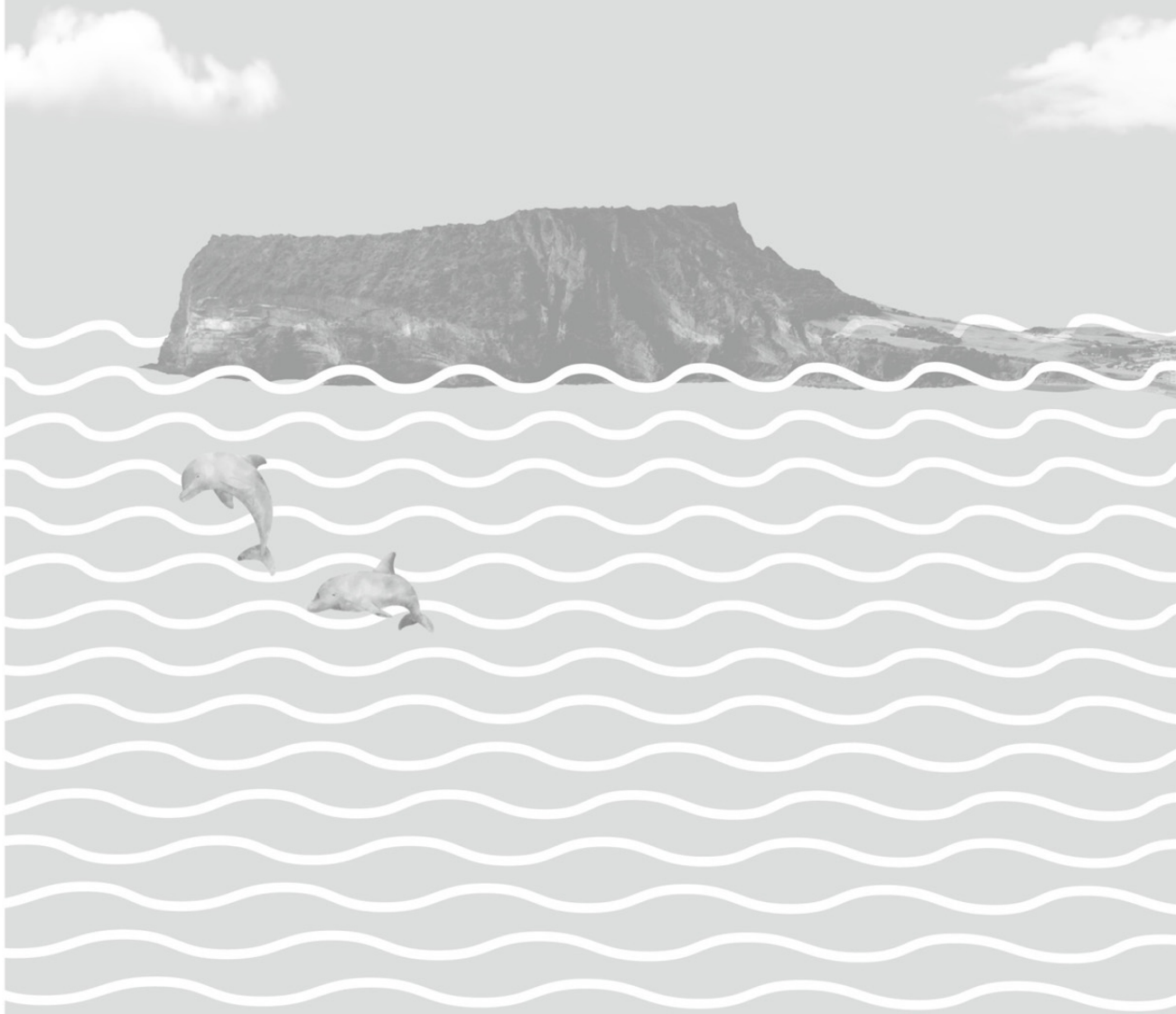
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Withdrawal

7분과

방사화학 (POSTER)



Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle

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This program aims to build a specialized and converged educational platform for the training of students in the back-end nuclear fuel cycle and cultivate integrated human resources encompassing majors, generations, and fields. To achieve this, we have established an infrastructure for integrated education and training in the radiochemistry and back-end nuclear fuel cycle and operated specialized educational courses linked with special lectures, experimental practices, and field trips.

Firstly, to construct an integrated educational and training infrastructure for the back-end nuclear fuel cycle, we formed a committee of experts from both inside and outside the institution and built an advanced radiochemistry laboratory equipped with physical and chemical analysis instruments. Through a comprehensive educational program involving theory, experiments, and discussions, we have established an integrated curriculum across adjacent majors and interdisciplinary studies. We also operate short-term education and experimental training programs (e.g., summer and winter schools for the back-end nuclear fuel cycle).

Secondly, the program has connected leading researchers domestically and internationally, as well as the next generation of scholars. The program offers long-term educational opportunities and internships targeting both undergraduate and graduate students. To support this, we continuously offer expert colloquiums and individual research internships. Through regular committee meetings and workshops, we focus on nurturing the integrated talents necessary for the back-end nuclear fuel cycle.

Through this program, students from various fields are being trained as competent integrated human resources capable of addressing various issues in the back-end nuclear fuel cycle. It is expected that this will enable us to supply specialized technical personnel in the back-end nuclear field in line with mid-to-long-term demands.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government Ministry of Science and ICT (MSIT) (No. 2021M2C7A1A02076339).

Keywords: Human resources development, Back-end nuclear fuel cycle, Radiochemistry education

Evaluation of Spectroscopic Methods for Determination of Hydrogen Peroxide as a Radiolysis Product

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Radiation from spent nuclear fuel (SNF) is one of key factors affecting the dissolution process of SNF and the source term from repository. The dissolution rate of uranium dioxide (UO₂) matrix of SNF is expected to control the release of radionuclides from SNF in contact with water under geological disposal conditions. Based on the oxidative dissolution mechanism, the solubility of UO₂ can increase significantly if the reducing environment near the fuel surface is altered by water radiolysis caused by radiation from SNF. Therefore, the analysis of water radiolysis products such as radicals ($\cdot\text{OH}$, $\cdot\text{OH}_2$, e_{aq} , $\cdot\text{H}$) and molecules (H_3O^+ , H_2 , H_2O_2) is prerequisite for studies on the rate of such dissolution process to determine oxidation/dissolution mechanism and related rate constants. In this study we examined the two-known spectroscopic methods developed for H₂O₂ determination; one is the luminol-based chemiluminescence (luminol-CL) method and the other is the spectrophotometry using ferrous oxidation-xylenol orange complexation (FOX). Their applicability for quantitative analysis of H₂O₂ in potential aqueous samples from SNF dissolution studies was evaluated in terms of the analytical dynamic range (ADR), the limit of detection (LOD) and the interfering effects of various metal ions possibly present in real samples.

The luminol-CL method exploits the chemiluminescence reaction caused by luminol; when in the presence of a metallic catalyst (e.g., Cu^{2+} , Co^{2+}), luminol emits a blue light (425 nm) at pH 10–11 in response to oxidizing agents such as hydrogen peroxide. Although a flow-through reaction system is routinely employed to enhance the analytical sensitivity we achieved the ADR up to $\sim 200\ \mu\text{M}$ and $\text{LOD} < 1\ \mu\text{M}$ by a batch-wise CL detection using conventional cuvette cells and an intensified charge-coupled device (ICCD). Interestingly, it turned out that the interfering effects of other metal ions (e.g., UO_2^{2+} , U^{4+} , Fe^{2+} and Fe^{3+}) is minimal, which should be advantageous for the luminol-CL method to be employed for samples potentially containing other metal ions.

On the other hand, the FOX method spectrophotometrically analyzes H₂O₂ based on the difference in color (or absorption spectra) of Fe-xylenol orange (XO) complexes. Initially, the Fe^{2+} -XO complex was provided in working solutions at pH 3, which was subsequently mixed with samples having H₂O₂ and allowed for quantitative oxidation of Fe^{2+} to Fe^{3+} . Typically, by monitoring the absorbance of Fe^{3+} -XO complex at 560–580 nm (λ_{max}) the ADR up to $\sim 100\ \mu\text{M}$ and $\text{LOD} \sim 1.6\ \mu\text{M}$ were achieved. However, it is found that interfering effects from M^{3+} and M^{4+} ions are significant; these interfering metal ions can form XO complexes so as to directly contribute the measured absorbance. In contrast, the influence from M^{2+} ions was found to be negligible.

To summarize we conclude that both methods can be applied for H₂O₂ determination for aqueous samples taken from SNF dissolution tests. However, prior to applying the FOX method the metal ion composition in those samples should be thoroughly identified not to overestimate the H₂O₂ concentration of samples. More details of underlying chemical reactions in both methods will be discussed in the presentation.

Keywords: Hydrogen peroxide, Radiolysis, Chemiluminescence, Luminol, Ferrous oxidation–xylenol orange complex

Determination of Inorganic Anions in Underground Water Samples With Capillary Electrophoresis by Using UV-Detection

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This study demonstrated a rapid and simple method for the determination of seven anions including halides and oxyhalides from the KURT underground water sample by capillary electrophoresis with UV detection. In nuclear waste disposal, some anions such as iodine, selenium, and technetium have been of great concern due to its high mobility and toxicity with a long half-life. It has been needed for a reliable analysis of anionic speciation because the high mobility of anions is easily affected by environmental conditions especially pH and salinity of underground water. Here this project is to develop a fast separation of seven anions including iodide, iodate, and selenite using capillary electrophoresis. The electroosmotic flow (EOF) was suppressed using a poly (ethyleneglycol)-coated capillary (DB-WAX capillary). As a result, anions migrated depending on their mobility under a reverse polarity condition (-15 kV) and the analysis time was within 15 min. UV detection was used at 200 nm. The RSDs for migration time were between 0.7% and 1.3% except for selenite of 5.1%. The RSDs for peak area were obtained between 2.9% and 7.4%. The calibration curves were linear from 10 to 200 mg/L with correlation coefficients greater than 0.9952. The LODs were 7.3, 10.9, 11.3, 12.9, 13.0, 13.9, and 17.4 mg/L for iodide, nitrate, bromide, selenite, bromate, tungstate and iodate. The KURT underground water sample spiked with seven anions at 50 mg/L were analyzed. The recoveries of spiked KURT sample ranged from 93.4% to 99.3%. The proposed method was successfully applied to determine seven anions in underground water sample.

Acknowledgements

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Keywords: Capillary electrophoresis, Analysis, Iodine, Selenium, High mobility

A Study on the pH Speciation and Stability Constants for Chelating Agents With Metals

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Chelating agents, such as ethylenediaminetetraacetic acid (EDTA), diethylenetriaminepentaacetic acid (DTPA), and nitrilotriacetic acid (NTA) are widely used in industry and agriculture as water softeners, detergents, and metal chelating agents. In wastewater treatment plants, a significant amount of chelating agents can be discharged into natural waters because they are difficult to degrade. Since those compounds affect the mobility of radionuclides or heavy metals in decontamination operations at nuclear facilities and radioactive waste disposal, quantification of the amount of ligand is very important for safe nuclear waste management. To predict the behavior of the main complexation in sample matrices of radioactive wastes, it is essential to evaluate the distribution of the metal-chelating species and their stabilities in order to develop analytical techniques for quantifying chelating agents.

We have investigated to collect information on the pH speciation of metal chelation and the stability constants of metal complexes depending on three chelating agents (EDTA, DTPA, and NTA). For example, Zhang's group recently reported that the initial coordination pH of Cu(II) and EDTA^{4-} is delayed with the addition of Fe(III), and the pH range for the stable existence of $[\text{Cu}(\text{EDTA})]^{2-}$ is narrowed compared to when it is alone in the sample matrix. The addition of Fe(III) clearly impacts the chemical states of the Cu(II)-EDTA solution. Additionally, Eivazihollagh's group demonstrated differences in the speciation and stability of Cu(II) species between Cu(II) and three chelating ligands (EDTA, DTPA, and NTA). This study will be greatly helpful in identifying the sample matrix for binding major chelating agents and metals as well as developing chemically sample pretreatment and separation methods based on the sample matrix. Finally, these advancements will enable reliable quantitative analysis of chelating agents in decommissioning radioactive wastes.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT) (RS-2023-00248338).

Keywords: Chelating agent, pH-speciation, Stabilization constants

Separation of Sr, Cs and Ba Using Gas Pressurized Extraction Chromatography With Cation Exchange Resin

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During the initial cooling period of spent nuclear fuel, Cs-137 and Sr-90 constitute a large portion of the total decay heat. Therefore, separating cesium and strontium from spent nuclear fuel can significantly decrease decay heat and facilitate disposition. This study presents analytical technique based on the gas pressurized extraction chromatography (GPEC) system with cation exchange resin for the separation of Sr, Cs, and Ba. GPEC is a micro-scaled column chromatography system that allows for faster separation and reduction volume of elution solvent compared to conventional column chromatography by utilizing pressurized nitrogen gas. Here, we demonstrate the comparative study of the conventional column chromatography and the GPEC method.

Cation exchange resin AG 50W-X12 (200~400 mesh size) was used. The sample was prepared at a 0.8 M hydrochloric acid solution and gradient elution was applied. In this case, we used the natural isotopes ⁸⁸Sr, ¹³³Cs, and ¹³⁸Ba instead of radioactive isotopes for the preliminary test. Usually, cesium is difficult to measure with ICP-OES, because its wavelengths (455.531 nm and 459.320 nm) are less sensitive. So, we used ICP-MS to determine the identification and the recovery of eluate. In this study, optimized experimental conditions and analytical result including reproducibility of the recovery, total analysis time and volume of eluents will be discussed by comparing GPEC and conventional column chromatography.

Keywords: Gas pressurized extraction chromatography, Column chromatography, Cation exchange resin, Strontium, Cesium, Barium

Establishment of Infrastructure for the Characterization of Intermediate-Level Waste and the Goal of KRID

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In Korea, extensive industry-academia-research research has already established many facilities and technologies for materials and chemical experiments on non-radioactive substances. However, few facilities have been built to analyze the physical and chemical properties of substances irradiated through neutron irradiation. Korea is planning to decommission Kori-1 and Wolsong-1 in 2027. Extensive analysis of low-level and intermediate-level materials is required to begin decommissioning these nuclear power plants. The material's composition and level can be identified by analyzing the structure's characteristics, and a cutting and decontamination plan can be established based on this. In addition, by conducting a nuclide analysis on the waste generated after cutting, suitability for disposal can be secured, and stable treatment can be performed. Accordingly, the Korea Decommissioning Research Institute (KRID) plans to secure infrastructure (hot cells) to analyze the characteristics of intermediate-level decommissioning waste. The goal is to secure high-dose/high-radiation decommissioning waste processing technology through Korea's first intermediate-level hot cell, support domestic nuclear power plant decommissioning projects, and secure and verify procedures related to nuclide analysis of intermediate-level using hot cells. In addition, by possessing these material properties and nuclide analysis technology, KRID can have a foundation to conduct continuous research on low- and intermediate-level radioactive materials and decommissioning. The purpose of KRID's establishment is to use this foundation in the future to improve the technological level of the nuclear industry as a whole through linkage between industry, academia, and research institutes.

Keywords: Intermediate-level radioactive material, Decommissioning waste, Hot cell, KRID

Determining the Optimal Conditions for the Sample Quantity of Radioactive Sludge

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Typically, the bottom of the effluent treatment facility at a nuclear power plant contains sediment, which is low-contamination waste consisting of sludge, gravel, sand, and other materials from which radioactive contaminants have been removed. Among these sediments, sludge is an irregular solid form consisting of small particles that are coagulated together, with radioactive isotopes containing cobalt attached.

Currently, there is a record of disposing of dry active waste from domestic nuclear power plants, and efforts are underway to gather basic data for the disposal of untreated waste such as sludge, spent filter, and spent resin. In particular, the classification and disposal methods of waste will be determined based on the radioactivity concentration. Therefore, plans are being made to determine the radioactivity concentration of radioactive isotopes and establish disposal plans for sludge samples.

In this study, pre-treatment and solutionization were carried out for the analysis of radioactive isotopes in sludge samples from nuclear power plants. The deviation of the gamma radioisotope analysis results was derived to obtain an optimal sample quantity that represents the sludge.

Keywords: Sludge, Radioactive waste, HPGe

Evaluation of Gross Alpha Radioactivity of Radioactive Sludge Samples

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In order to establish disposal plans for sludge, which is one of the untreated waste materials from domestic nuclear power plants, it is necessary to determine the radioactivity concentration of radioactive isotopes. In this study, we aim to evaluate the gross alpha radioactivity of sludge containing radioactive contaminants after pre-treatment, in order to assess the level of sludge waste and obtain analytical data for discussing disposal methods. Samples of sludge generated from nuclear power plants were pre-treated, solutionized, and prepared as analysis samples for evaluating the gross alpha radioactivity.

Keywords: Sludge, Radioactive waste, Gross alpha

Sorption Characteristics of Se(IV) in Clay Mineral Suspensions (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle)

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The ultimate objective of deep geological repositories is to achieve complete segregation of hazardous radioactive waste from the biosphere. Thus, given the possibility of leaks in the distant future, it is crucial to evaluate the capability of clay minerals to fulfill their promising role as both engineered and natural barriers. Selenium-79, a long-lived fission product originating from uranium-235, holds significant importance due to its high mobility resulting from the predominant anionic form of selenium. To investigate the retardation behaviors of Se(IV) in clay media by sorption, a series of batch sorption experiments were conducted. The batch samples consisted of Se(IV) ions dissolved in 0.1 M NaCl solutions, along with clay minerals including kaolinite, montmorillonite, and illite-smectite mixed layers. The pH of the samples was also varied, reflecting the shift in the predominant selenium species from selenious acid to selenite ion as the environment can shift from slightly acidic to alkaline conditions. This alteration in pH concurrently promotes the competition of hydroxide ions for Se(IV) sorption on the mineral surface as the pH increases and impedes the selective attachment of selenium. The acquired experimental data were fitted through Langmuir and Freundlich sorption isotherms. From the Freundlich fit data, the distribution coefficient values of Se(IV) for kaolinite, montmorillonite, and illite-smectite mixed layer were derived, which exhibited a clear decrease from 91, 110, 62 L/kg at a pH of 3.2 to 16, 6.3, 12 L/kg at a pH of 7.5, respectively. These values derived over the pH range provide quantitative guidance essential for the safety assessment of clay mineral barriers, contributing to a more informed site selection process for deep geological repositories.

Acknowledgements

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Keywords: Deep geological repository, High level radioactive waste, Selenium, Clay minerals, Sorption

Electrochemical Behaviors of UCl_3 and CrCl_2 in NaCl-MgCl_2 Salt (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle)

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Molten chloride salts have received considerable research attention as potential nuclear fuel and coolant candidates for molten salt reactors. However, there are several challenges, especially for structural materials due to the selective dissolution of chromium (Cr) in the molten chloride salts environment. Understanding the compatibility of uranium (U), which is used as nuclear fuel in molten salt reactors, with Cr in molten chloride salts is critical for designing the molten salt reactor structure. Therefore, in this study, the cyclic voltammetry (CV) was used to investigate the electrochemical behaviors of U and Cr. The diffusion coefficients and formal potentials were obtained.

The electrochemical properties of uranium and chromium were investigated by CV in molten NaCl-MgCl_2 salt at 600°C . Tungsten rods for working and counter electrode, and Ag/AgCl for reference electrode were utilized in this experiment. UCl_3 made from the chemical dissolution of U rods and CrCl_2 (Sigma-Aldrich, 99.99%) were used.

Diffusion coefficients (D) of U and Cr were calculated by measuring reduction peak current of U^{3+}/U and Cr^{2+}/Cr from CV curves and using the Berzins-Delahay equation; $D(\text{U}^{3+}/\text{U}) = 3.0 \times 10^{-5} \text{ cm}^2 \text{ s}^{-1}$ and $D(\text{Cr}^{2+}/\text{Cr}) = 3.3 \times 10^{-5} \text{ cm}^2 \text{ s}^{-1}$. The formal potentials were also calculated by using the reduction peak potential obtained from CV results; $E^{0'}(\text{U}^{3+}/\text{U}) = -1.173 \text{ V}$ and $E^{0'}(\text{Cr}^{2+}/\text{Cr}) = -0.321 \text{ V}$. The ionization tendency was investigated by comparing each reduction peak potential. The reduction peak potential $E_{p,c}$ was increasing order of $E_{p,c}(\text{U}^{3+}/\text{U}) < E_{p,c}(\text{Cr}^{2+}/\text{Cr}) < E_{p,c}(\text{U}^{4+}/\text{U}^{3+})$. It can be seen that in the presence of U^{4+} and Cr metals, the Cr in the alloy can dissolve into Cr^{2+} , but in the presence of U^{3+} and Cr metals, the Cr in the alloy does not dissolve into Cr^{2+} .

By analyzing the CV curve, diffusion coefficients and formal standard potentials were obtained. The result of comparing reduction peak potentials suggests that the nuclear fuel using U^{4+} should be inhibited to prevent the selective dissolution of Cr.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government Ministry of Science and ICT (MSIT) (No. 2021M2C7A1A02076339).

Keywords: Molten salt, Cyclic voltammetry, Diffusion coefficients, Ionization tendency

Application of Laboratory-Based XAFS Equipment for Radiochemical Analysis (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle)

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A comprehensive understanding of actinide coordination chemistry and its structure is essential in many aspects of the nuclear fuel cycle, such as fuel reprocessing, waste management, reactor safety, and non-proliferation efforts. Managing radioactive waste generated during the nuclear fuel cycle has recently become more important, accordingly increasing the importance of designing appropriate waste forms and storage solutions for long-term waste disposal. Compared to the increase in the need for understanding the chemistry of major radioactive elements, the information on the local structure of the radioactive elements, especially actinides, remains unknown. To probe this issue, X-ray absorption fine structure (XAFS) can be applied. By analyzing the EXAFS (extended X-ray absorption fine structure) and XANES (X-ray absorption near edge structure), the local structure around atoms can be determined. However, the radioactive properties of the nuclides hindered the measurement of EXAFS and XANES, due to the difficulties of preparation, containment, and transfer of the sample.

To measure the EXAFS of various compounds regarding the back-end nuclear fuel cycle, laboratory-based EXAFS (hiXAS, HP spectroscopy) has been introduced which can measure the EXAFS and XANES at the energy range of 5-18 keV. Compounds of Copper (Cu foil, CuO samples), Zirconium (Zr foil), and Europium (Eu₂O₃) were used for the verification of the laboratory-based EXAFS at a given energy range. The measured EXAFS spectrum of various compounds exhibit good agreement with the theoretical data, showing an R-factor of less than 0.02. It was found that each graph has a first peak corresponding to 2.55 Å for Cu foil (Cu-Cu), 1.93 Å for CuO samples (Cu-O), 3.23 Å for Zr foil (Zr-Zr), and from 2.32 Å to 2.34 Å for Eu₂O₃ (Eu-O), which agree well with other values from the literature. From the result, it can be implied that this equipment can be used especially in the back-end nuclear fuel cycle field to enhance the understanding of local structure in radiochemistry.

Acknowledgements

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Keywords: X-ray absorption spectroscopy, Local structure, EXAFS/XANES

Kinetic and Isothermal Studies of Uranium Adsorption on Phosphate-Functionalized Mesoporous Silica Microsphere Composite

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Bis (2-ethylhexyl)phosphoric acid (HDEHP) is a renowned extractant, favored for its affinity to selectively remove uranium via its P=O groups. We previously synthesized HDEHP-functionalized mesoporous silica microspheres for solid-phase uranium adsorption. Herein, we investigated the kinetic and isothermal behavior of uranyl ion adsorption in mesoporous silica microspheres functionalized with phosphate groups. Adsorption experiments were conducted by equilibrating 20 mg of silica samples with 50 mL of uranium solutions, with concentrations ranging from 10 to 100 mgU L⁻¹ for isotherms and 100 mgU L⁻¹ for kinetics. Three distinct samples were prepared with varying HDEHP to TEOS molar ratios ($x = 0.16$ and 0.24) and underwent hydrothermal treatment at different temperatures, resulting in distinct textural properties. Contact times spanned from 1 to 120 hours.

For $x = 0.16$ samples, it took around 50 and 11 hours to reach equilibrium for the hydrothermally treated samples at 343 K and 373 K, respectively. Adsorbed quantities were similar (99 and 101 mg g⁻¹, respectively), indicating consistent functional group content. This suggests that the key factor influencing uranium adsorption kinetics is pore size of the silica. The sample treated at 373 K, with a larger pore size (22.7 nm) compared to 343 K (11.5 nm), experienced less steric hindrance, allowing uranium species to diffuse more easily through the mesopores. The data confirmed the excellent fit of pseudo-second-order kinetic model ($R^2 > 0.999$) and closely matched the experimental value, suggesting that chemisorption governs the rate-controlling step.

To gain further insights into uranium adsorption behavior, we conducted an adsorption isotherm analysis at various initial concentrations under a constant pH of 4. Both the Langmuir and Freundlich isotherm models were applied, with the Langmuir model providing a superior fit. The relatively high R^2 value indicated its effectiveness in describing the adsorption process, suggesting homogenous sorbate adsorption on an energetically uniform adsorbent surface via a monolayer adsorption and constant adsorption site density, without any interaction between adsorbates on adjacent sites. Remarkably, differences in surface area did not significantly impact uranium removal efficiency. This observation strongly suggests that the adsorption capacity is primarily governed by the loading amount of HDEHP and the inner-sphere complexation with the phosphoryl group (O=P). Our silica composite exhibited an impressive adsorption capacity of 133 mg g⁻¹, surpassing the results reported in the majority of other silica literature.

Keywords: Silica, Uranium, Adsorption, Kinetics, Isotherms

Evaluation of Density and Coincidence Summing Correction on Low-Level Concrete Waste for Gamma Spectroscopy

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Concrete is the primary building material for nuclear facilities, making it one of the most common forms of radioactive waste generated when decommissioning a nuclear facility. Of the total waste generated at the Connecticut Yankee and Maine Yankee nuclear power plants in the United States, concrete waste accounts for 83.5% of the total for Connecticut Yankee and 52% for Maine Yankee. In order to dispose of the low- to medium-level radioactive concrete waste generated during the decommissioning of nuclear power plants, it is necessary to analyze the radioactivity concentration of gamma nuclides such as Co-58, Co-60, Cs-137, and Ce-144.

Gamma-ray spectroscopy is commonly used method to measure the radioactivity concentration of gamma nuclides in the radioactive waste; however, due to the nature of gamma detectors, gamma rays from sequentially decaying nuclides such as Co-60 or Y-88 are subject to True Coincidence Summing (TCS). TCS reduces the Full Energy Peak Efficiency (FEPE) of specific gamma ray and it can cause underestimation of radioactivity concentration. Therefore the TCS effect must be compensated for in order to accurately assess the radioactivity of the sample. In addition, samples with high density and large volume will experience a certain level of self-shielding effect of gamma rays, so this must also be compensated for.

The Radioactive Waste Chemical Analysis Center at the Korea Atomic Energy Research Institute performs nuclide analysis for the final disposal of low- and intermediate-level concrete waste. Since a large number of samples must be analyzed within the facility, the analytical method must simultaneously satisfy accuracy and speed. In this study, we report on the results of evaluating the accuracy of the radioactivity concentration correction by applying an efficiency transfer method that appears to satisfy these requirements to concrete standard reference material.

Keywords: HPGe, LVIS, Coincidence summing, Material density, Radioactive waste, Concrete, Gamma spectroscopy

Necessity for Chelate Analysis Technology in Nuclear Decommissioning Waste

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As existing nuclear power plants reach the end of their lifespan, 22 nuclear power plants in Korea are scheduled to be permanently shut down and decommissioned by 2050. Chelates are used as decontamination agents during nuclear power plant operation and decommissioning, and as a result, decommissioning waste contains chelates. Chelates contained in radioactive waste are complexed with radionuclides and increase their mobility. So, qualitative and quantitative analysis of chelates contained in radioactive waste is necessary. However, the spectroscopic method (UV-Vis), previously used for chelate analysis in Korea takes too much time for analysis and cannot analyze two or more chemically similar chelates at the same time. Due to these problems, new methods for analyzing chelate must be developed. Overseas, many cases of chelate analysis using advanced analysis equipment have been reported. CEA in France has developed a chelate analysis method for application to radioactive waste using HPLC-MS (J. Chromatogram. A, 1276, 20-25, 2013). In this method, the existing method of measuring EDTA using a complex of Fe and EDTA was improved to measuring a complex of Ni and EDTA. Based on such overseas cases, we would like to develop an analysis method for chelates in radioactive waste. For this purpose, we will verify similar overseas papers and develop pretreatment methods for mixtures of chelates (EDTA, DTPA, NTA) and metals (Fe, Ni, Cu, etc.) in various media. Finally, we will develop a separation analysis technology for multi-component chelates in nuclear decommissioning waste based on LC-MS.

Acknowledgements

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Keywords: Chelate, EDTA, DTPA, NTA, LC-MS, Nuclear decommissioning waste

Determination of EDTA, DTPA, and NTA in Nuclear Decommissioning Waste by Liquid Chromatography-Mass Spectrometry

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Chelating agents in low and intermediate radioactive wastes can form complexes with radionuclides and increase the mobility of the radionuclides. According to the Korea Radioactive Waste Agency (Acceptance criteria for low and intermediate radioactive waste, WAC-SIL-2022-1), if the amount of residual chelating agents in the waste are greater than 0.1%, the chemical names and residual amounts should be specified; if greater than 1%, the waste must be solidified and contain no more than 8%. The existing method for analyzing chelates in radioactive waste was based on UV-Visible spectrophotometry (UV-Vis), but the new method is based on liquid chromatography/mass spectrometry (LC-MS). The analysis was performed in aqueous solution before applying to real samples. Since the real sample may contain several heavy metals, it is expected that the chelates will exist as complexes. Therefore, 1.0×10^{-4} mol L⁻¹ of EDTA (Ethylenediaminetetraacetic acid), DTPA (Diethylenetriaminepentaacetic acid), NTA (Nitrilotriacetic acid), and excess metals in aqueous solution were analyzed using HPLC using RP (Reverse Phase) column and HILIC (Hydrophilic interaction) column. When the RP column was used, each substance eluted without separation at the beginning of the analysis. However, when analyzed using a HILIC column, the peaks of each substance were separated. LC-MS measurements using HILIC conditions resulted in separations with better sensitivity.

Acknowledgements

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Keywords: EDTA, DTPA, NTA, LC-MS, HILIC, Nuclear decommissioning waste

Comparison and Evaluation of Pre-Treatment Methods for Chemical Separation of Ni-59, Ni-63 and Fe-55 Difficult-to-Measure Radionuclides in Decommissioning Waste

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In the decommissioning process of nuclear power plants, Ni-59, Ni-63 and Fe-55 present in radioactive waste are crucial radionuclides used as fundamental indicators in determining waste treatment methods. However, due to their low-energy 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 goal is to find the most optimized pre-treatment method for the analysis of Ni-59, Ni-63 and Fe-55 in decommissioning waste.

The comparative evaluation results confirm that the chemical separation procedures for Fe and Ni 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.

Keywords: Radioactivity analysis, Difficult-to-measure, Fe-55, Ni-63, Ni-59

Improved Algorithm for Gamma- and X-spectroscopic Analysis of Iodine-129 Considering of Energy-Efficiency Correation

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I-129 is one of the imporant nuclides that must be determined in the disposal process of radioactive waste in many countries. This radionuclide emits gamma-ray and x-ray photons within the energy range of 29 to 39 keV, consequently, an x-ray detector with high resolution performance is required for the analysis of I-129 activity. An n-type coaxial HPGe detector exhibits higher efficiency characteristics compared to a planar-type HPGe detector, however, its resolution is lower than a planar type. So it is difficult to completely deconvolute and fit the gamma-ray and x-ray peaks in the spectrum using a general gamma-ray spectrum analysis program such as GammaVision. To address this problem, in a previous study introduced the developed algorithm for the fitting and analysis of I-129 gamma-ray and x-ray spectrum by fixing their emission ratios. In this study, we improved the algorithm by considering the variation of the efficiency in the HPGe spectrum, which reflects the actual HPGe crystal condition. And algorithm tests were performed using measured I-129 sample spectra with interfering nuclides acting as background curve are introduced.

Keywords: Fitting algorithm, I-129, X-ray, HPGe detector

Comparative Study on Direct and Indirect ^{63}Ni Analysis Method in Radioactive Waste

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For safe disposal of radioactive wastes, accurate analysis of nuclear isotopes is important. It is known that there are 14 nuclides that have to identify nuclide-specific concentration levels. ^{63}Ni , one of non-volatile nuclear isotopes which is included in those 14 nuclides, has to follow chemical separation for exact analysis.

As various analysis methods were developed, various methods for analyzing ^{63}Ni also emerged. Past method has used measurement specimens of ^{59}Ni , after ^{59}Ni measurement has been done. It used HClO_4 , known as strong oxidizing agent, to dissolve DMG, an organic substance used to form ^{59}Ni precipitates. Nowadays, we analyze ^{59}Ni and ^{63}Ni simultaneously, which enables short analysis time, without use of HClO_4 . But high accuracy is just as important as short measurement time and efficiency. So, this paper compare ^{63}Ni specific activity value used new method with the value, past method used, using real sample's data.

As a result, all sample data from new method's relative ^{63}Ni specific activity is within the uncertainty range of past ones based on past specific activity value. Consistency of new method's result and past method's data increased the reliability of the data and accuracy of those methods.

Keywords: ^{63}Ni , ^{59}Ni , Specific activity value, HClO_4 , DMG

The Separation Method of ^{90}Sr and ^{99}Tc by a Carbonate Precipitation

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Currently, non-volatile nuclides such as ^{94}Nb , ^{99}Tc , ^{90}Sr , ^{55}Fe , and $^{59/63}\text{Ni}$ are used a sequential separation. In this study, we developed a separation for ^{99}Tc and ^{90}Sr by a carbonate precipitation.

Sodium Carbonate (Na_2CO_3) was inserted in the aqueous sample from a Dry Active Waste (DAW) and a carbonate precipitation was produced. The precipitate is composed of di- or tri-valent element such as Co, Sr, Fe, Ni and the supernatant is composed of mono-valent element (Cs) and anion materials (ReO_4^- , TcO_4^-). In DAW, it was confirmed that the recovery of ^{90}Sr (precipitate) and ^{99}Tc (supernatant) were $> 90\%$, respectively. The precipitate and supernatant separated by using a Sr-resin and an anion-exchange resin, respectively. The separated samples were measured by a Liquid Scintillation Counter (LSC, ^{90}Sr) and Inductively-Coupled Plasma-Mass Spectroscopy (ICP-MS, ^{99}Tc).

Keywords: Separation, ^{90}Sr , ^{99}Tc

Pre-Test on Radioactivity of Steel Materials

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In this study, we introduce the validation of the analysis guidelines through preliminary experiments of the draft analysis guidelines before analyzing waste materials (non-combustible). This validation data was applied the accuracy and efficiency of the separation and analysis for the waste such as steel generated from NPP.

Steel (non-flammable) was leached the mixed acid and the leaching solution was separated by using the separation guidelines. Steel was corroded with radioactive RM (Co-60, Cs-137) and mixed acid. After drying, the corroded steel was measured the initial radioactivity by a HPGe detector (10,000 seconds). The sample was inserted in a beaker and leached with mixed acid (10 M HNO_3 + 4 M HCl) for 2 hours. In this solution, it added 2 ml of H_2O_2 to increase the leaching effect. The ultrasonic device was adjusted so that the temperature does not exceed 60°C. After elution, the surface of the sample was washed with pure water. The weight of the sample was measured accurately, and recorded the weight loss rate after infiltration. The leaching sample was measured radioactivity by a HPGe detector (10,000 seconds). It was calculated the recovery rate based on the difference in total radioactivity before and after leaching.

Before the test, radioactive RM (Co-60, Cs-137) was radioactive deposited by corrosion, but Cs-137 was not detected in the initial gamma measurement and only Co-60 nuclides were deposited. The recovery rate test results were confirmed to be about 100%.

Keywords: Steel, Co-60, Cs-137, Recovery test

Removal of Oxides in NaCl-MgCl₂ by Thermodynamic and Electrochemical Consideration for Molten Salt Reactor

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Molten Salt Reactor (MSR) is one of the 4th generation nuclear power systems which is its verified technology in physically and chemically. Among the various salts used for MSR system, the eutectic composition of NaCl-MgCl₂ system maintains the liquid state at around 450°C, in the same time, it has high solubility for nuclear fuel chlorides. This characteristic has high advantage for lowering the operating temperature for the MSR, which could reduce the problem of high-temperature corrosion by salt for structural materials significantly. In particular, since MgCl₂ has the similar standard reduction potential with nuclear fuel, is used as a surrogate for, many basic researches have been conducted for verifying characteristic of MgCl₂. It is well-known that main short-advantage of MgCl₂ is hygroscopic properties. MgCl₂ changes to MgCl₂·xH₂O state easily by absorbing moisture in air condition. The hydrated MgCl₂ is producing MgOHCl by thermally decomposing at high temperature, the formed MgOHCl corrodes structural materials, even small amount of MgOHCl gives significant damage. Therefore, the purification of MgCl₂ has been required for long-term operation of MSR using MgCl₂ as a base salt. In this study, the purification of eutectic composition salt for NaCl-MgCl₂ has been mainly performed by considering its thermodynamic properties and electrochemical characteristic, and the experimental results have been discussed.

Keywords: Molten salt reactor, NaCl-MgCl₂, Corrosion, Salt purification

A New Method Using UV Irradiation for Radiocarbon Analysis

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Most of the C-14 produced is in the organic form, generated as methane ($^{14}\text{CH}_4$), methanol ($^{14}\text{CH}_3\text{OH}$), formaldehyde ($^{14}\text{CH}_2\text{O}$), and formic acid ($^{14}\text{CO}_2\text{H}_2$). When analyzing C-14, it is transformed into the form of $^{14}\text{CO}_2$, and its concentration is determined using LSC. Typical examples include the wet oxidation method, the combustion or Pyrolysis. The wet oxidation method uses strong acids and involves repeated operations, which generates large amounts of acid waste and secondary radioactive waste. The combustion method uses high temperatures, which requires an oxygen device. Pyrolysis also requires high temperature in a vacuum and catalysts. Catalysts are expensive because they are platinum-based. To compensate for these shortcomings, a C-14 analysis method using UV irradiation was developed. In this study, 100 mL of distilled water mixed with formic acid (CO_2H_2), potassium persulfate ($\text{K}_2\text{S}_2\text{O}_8$), and silver nitrate (AgNO_3) was irradiated with a 320-390 nm UV lamp to conduct a CO_2 production reaction experiment. The UV range was measured using a photometer (UV Power puck II). The beaker was made of quartz in 150 mL size with three inlets : a temperature measurement, a sample inlet, and a collection tube connector. We changed the UV lamp used from a 450 W halogen lamp to a 100 W LED, which has a lower temperature and is safer. As a result of the experiment, CO_2 bubbles were generated in the collection tube, due to the UV irradiation react, which uses oxidizer and catalysts. The maximum temperature of the solution irradiated with the LED UV lamp was less than 56°C . It confirmed the rate of bubble generation changed depending on the lamp distance, the amount of sample, oxidizer, and catalyst. In an experiment to confirm the reaction caused by heat, it was found that although a reaction occurred due to heat, the reaction was significantly lower than when using a UV lamp. The reproducibility experiment was conducted three times in total under the same conditions. It showed the same pattern. In the future, we plan to select mock samples, collect $^{14}\text{CO}_2$ in Carbo-Sorb, and analyze them using LSC. The results of this research will be used as a technology to recover C-14 more safely and efficiently and will also be used to expand its application to the treatment of other wastes such as waste liquid and waste resin through simulated samples.

Keywords: Radiochemistry, Photochemical, Radiocarbon, Analysis, Radioactive waste

Sorption and Solidification of Radioactive Cobalt Using Hydroxyapatite and Geopolymer Composite (Program for Fostering Convergence of Human Resources in Back-end Nuclear Fuel Cycle)

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The need for the development of sustainable, efficient, and green radioactive waste disposal methods is emerging with the saturation of spent nuclear waste storage facilities in the Republic of Korea. Conventional radioactive waste management methods like using cement or glass have drawbacks such as high porosity, less chemical stability, high energy consumption, carbon dioxide production, and the generation of secondary wastes, etc. To address this gigantic issue of the planet, we have designed a study to explore the potential of alternative materials having easy processability, low carbon emissions and more chemical stability such as ceramic (hydroxyapatite, HAP) and alkali-activated materials (geopolymers, GP) to capture the simulated radioactive cobalt ions from the contaminated water and directly solidify them at low temperatures.

Physical and mechanical properties of HAP alone and 15wt% GP incorporated HAP (HAP-GP-15) composite were studied and compared. The surface of both materials was fully sorbed with an excess amount of Co(II) ions in the aqueous system. Co(II) sorbed powders were separated from aqueous media using a centrifuge machine operating at 5,000 RPM for 10 minutes and dried at 100°C for 8 hours. The dried powders were then placed in stainless steel molds, and shaped into cylindrical pellets using a uniaxial press at a pressure of 1 metric ton for 1 minute. The pellets were sintered at 1,100°C for 2 hours at a heating rate of 10°C/min. Following this, the water absorption, density, porosity, and compressive strength of the polished pellets were measured using standard methods. Results showed that HAP has a greater potential for decontamination and solidification of Co(II) due to its higher density ($2.65 \text{ g/cm}^3 > 1.90 \text{ g/cm}^3$), less open porosity ($16.2 \pm 2.9\% < 42.4 \pm 0.9\%$) and high compressive strength ($82.1 \pm 10.2 \text{ MPa} > 6.9 \pm 0.8 \text{ MPa}$) values at 1,100°C compared to that of HAP-GP-15. Nevertheless, further study with different constituent ratio of HAP and GP at various temperatures is required to fully optimize the HAP-GP matrix for waste solidifications.

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Keywords: Hydroxyapatite, Geopolymers, Sorption, Solidification, Radioactive waste

Application of Pyrolysis for the Analysis of Volatile Radionuclides in Activated Carbon

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Within the air purification system of a nuclear power plant, specific radioactive isotopes are extracted from gases through adsorption onto activated carbon. To properly dispose of used activated carbon, it is essential to determine the concentration of radioactive nuclides within it. This study discusses the application of the pyrolysis method for analyzing the concentrations of ^3H and ^{14}C in spent activated carbon. The pyrolysis was conducted using Raddec's Pyrolyser, with adjustments made to parameters such as temperature profiles, airflow rates, sample quantities, and trapping solution volumes. The evaluation method for the pyrolysis of activated carbon to analyze ^3H and ^{14}C involved adding ^3H and ^{14}C sources to the activated carbon before use and subsequently assessing the recovery rates of the added sources in comparison to the analysis results.

Keywords: ^3H , ^{14}C , Pyrolysis, Activated carbon

Planar Distribution of Noble Metal Phase in a Used Fuel Assembly

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Noble metal phase, present in used fuel, are fission products that can be found as metallic precipitates in used nuclear fuel. They exist as small particles (nm~um) in grain boundaries of the used fuels. Since they are particles deposited between the grain structures, they can be considered as defects in the pellet structure. Thermal expansion of fuels with noble metal is slightly higher than that of bare fuels. The fuels at high temperature, such as immediately after being discharged from nuclear reactors, may be subject to fuel failure if sufficient cooling is not provided. Recent research has shown that the noble metals can migrate into the rim space between the pellet and the cladding, and be deposited in the inner layer of the claddings. therefore, the mechanical integrity of the cladding can be degraded by noble metals, as well as the pellets.

The concentration of the noble metal phase should be considered to evaluate the effect of the noble metals on the fuel integrity, after discharge from the reactors. SCALE/ORIGEN code was used to evaluate the noble metals in fuel assembly-scale, and the radial distribution in the fuel assembly. The radial distribution of the reactor power was derived from the SCALE/TRITON, considering Westinghouse 17×17. Square cell model was chosen for the geometry and 1/4 model was applied to reduce the computation time.

Keywords: Used fuel, Noble metal phase, High-level waste, Grain boundaries, Fuel assembly

Preliminary Study of Fission Product Behavior in 100 Mw Chloride Based Molten Salt Reactor

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Molten Salt Reactor, which employs molten salt mixture as fuel, has many advantages in reactor size and operation compared to conventional nuclear reactor. In developing Molten Salt Reactor, the behavior of fission product in operation should be preliminary evaluated for the correct design of reactor and its associated system including off-gas treatment. In this study, for 100 Mw 46 KCl-54 UCl₃ based Molten Salt Reactor with operating life time of 20 year, the fission product behavior was estimated by thermodynamic modeling employing FactSage 8.2. Total inventory of all fission product were firstly calculated using OpenMC code allowing depletion during neutronic calculation. Then, among all inventory, 46 element species from Uranium to Holmium were chosen and given to the input for equilibrium module of Factsage with its mass. In phase equilibrium calculation, for the correct description of solution phase, KCl-UCl₃ solution database based on modified quasi-chemical model in the quadruplet approximation (ANL/CFCT-21/04) was employed and the coexisting solid phase was assumed to pure state. With the assumption of no oxygen and moisture ingress into reactor system, equilibrium calculation showed that 1% of solid phase and of gas phase were newly formed and, in gas phase, major species were identified : ZrCl₄ (47%), Xe (33%), UCl₄ (14%), Kr (5%), Ar (1%) and others. This result reveals that off-gas treatment of system should account for the appropriate treatment of ZrCl₄ and UCl₄ besides treatment of noble gas such as Xe and Kr.

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Keywords: Molten salt reactor, Chloride salt, Factsage, Equilibrium calculation, Fission product behavior

Quantitative Analytical Methodology of D₂O Content in Natural and Enriched Water

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Heavy water (deuterium oxide, D₂O) is water in which hydrogen atoms (¹H, H), one of the constituent elements of water molecules, have been replaced with deuterium (²H, D), a heavier isotope. Heavy water is used in a variety of industries, including semiconductors, nuclear magnetic resonance, infrared spectroscopy, neutron deceleration, neutrino detection, metabolic rate studies, neutron capture therapy, and the production of radioactive materials such as plutonium and tritium. In particular, heavy water is used as a neutron moderator or coolant in nuclear reactors and as a fuel for nuclear fusion energy, methods for measuring heavy water are becoming increasingly important. There are methods with density, mass spectrometry, and infrared (IR) spectroscopy.

In this study, Fourier transform infrared spectroscopy (FT-IR) was used, which is commonly used in IR spectroscopy because of its relatively high analytical sensitivity, low operating costs, and easy online analysis. Heavy water was identified in the range of 2,300-2,600 cm⁻¹ wavenumber (O-D) and the range of 1,200-1,300 cm⁻¹ wavenumber (D-O-D), which are known to be the range with strong infrared absorption. As a result, the linearity of infrared absorbance for each heavy water concentration was confirmed within the relative expansion uncertainty (k=2).

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Keywords: Heavy water, D₂O, Enriched water, FTIR

pH-Dependent Composition Analysis of Nickel-Chelate Complexes Using UV-Vis Spectrophotometry

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Chelating agents like ethylenediaminetetraacetic acid (EDTA), diethylenetriaminepentaacetic acid (DTPA), and nitrilotriacetic acid (NTA) find extensive application in the removal of residual substances due to their high stability constants with a wide range of metal ions. They also play a crucial role in nuclear decontamination operations aimed at eliminating metallic radionuclides such as ⁶⁰Co, ⁹⁰Sr, and ²³⁹Pu. However, improper disposal of chelated radioactive waste can lead to significant increases in radionuclide migration rates from disposal sites. Therefore, it is imperative to comprehend the behavior of chelating agents under varying conditions, including pH, temperature, and metal ion concentrations.

In this study, we present the results of a pH-dependent composition analysis of nickel-chelate complexes using UV-Vis spectrophotometry. Nickel (Ni) serves as an ideal metal ion for investigating its interactions with chelating agents due to its solubility over a wide pH range and high stability constants with all three chelating agents mentioned earlier. Initially, UV-Vis spectra of Ni-EDTA, Ni-DTPA, and Ni-NTA complexes were recorded at various pH levels. We assigned absorption maxima and compared our findings with existing literature on each Ni-chelate complex. Furthermore, we examined mixed samples of all three complexes, varying the pH to monitor changes in composition. The results and their implications will be presented in our poster presentation.

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Keywords: Chelating agents, Radioactive waste, Nickel, UV-Vis spectrophotometry

Surface Characterization of ϵ -particle, Trivalent- and Tetravalent-metal Doped Uranium Dioxide

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This study explores the impact of metal doping on the surface structure of spent nuclear fuels (SNFs), particularly uranium dioxide (UO_2). SNFs undergo significant microstructural changes during irradiation, affecting their physical and chemical properties. Certain elements, including actinides and lanthanides, can integrate into the UO_2 lattice, leading to non-stoichiometry based on their oxidation state and environmental conditions. These modifications are closely linked to phenomena like corrosion and oxidation of UO_2 , making it essential to thoroughly characterize SNFs influenced by specific element doping for disposal or interim storage decisions.

The research employs X-ray diffraction (XRD), scanning electron microscopy (SEM), and Raman spectroscopy to investigate the surface structure of UO_2 samples doped with elements such as Nd^{3+} , Gd^{3+} , Zr^{4+} , Th^{4+} , and ϵ -particles (Mo, Ru, Pd). To manufacture these samples, UO_2 powders are mixed and pelletized with the respective dopant oxide powders. The resulting pellet samples are sintered under specific conditions.

The XRD analysis reveals that the lattice parameters of $(\text{U,Nd})\text{O}_2$, $(\text{U,Gd})\text{O}_2$, $(\text{U,Zr})\text{O}_2$, and $(\text{U,Th})\text{O}_2$ linearly vary with increasing doping levels, suggesting the formation of solid solutions. SEM images show that the grain size decreases with higher doping levels in $(\text{U,Gd})\text{O}_2$, $(\text{U,Nd})\text{O}_2$, and $(\text{U,Zr})\text{O}_2$, while the change is less pronounced in $(\text{U,Th})\text{O}_2$. Raman spectroscopy uncovers that $\text{U}_{0.9}\text{Gd}_{0.1}\text{O}_{2-x}$ and $\text{U}_{0.9}\text{Nd}_{0.1}\text{O}_{2-x}$ exhibit defect structures related to oxygen vacancies, induced by trivalent elements replacing U^{4+} , distorting the UO_2 lattice. In contrast, $\text{U}_{0.9}\text{Zr}_{0.1}\text{O}_2$ shows no oxygen vacancy-related defects but features a distinct peak, likely indicating the formation of a ZrO_8 -type complex within the UO_2 lattice. ϵ -Particle doped uranium dioxide shows minimal deviations in surface properties compared to pure UO_2 .

This structural characterization of metal-doped and ϵ -particle-doped UO_2 enhances our understanding of spent nuclear fuel behavior, with implications for the characterization of radioactive materials. This research provides valuable insights into how specific element doping affects the properties of SNFs, which is crucial for managing and disposing of these materials safely.

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Keywords: Uranium dioxide, ϵ -particle, Solid solution, Surface structure

Infrared Radiation-Assisted Sensing in Chemisorption of Radioactive Organic Iodides Using Metal-Organic Frameworks

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Owing to the rapid rise of global energy demands, the operation of nuclear power plants is still indispensable. However, following the nuclear accident at Fukushima-Daiichi in 2011, the secure sequestration of radioactive waste has become critical for ensuring safe operations. Among various forms of nuclear wastes, capturing radioactive organic iodide (ROIs, e.g., methyl iodide, ethyl iodide, and propyl iodide) as one of the important species in gas phase waste has been challenged owing to the insufficient sorbent materials. The environmental release of ROIs with high volatility can give rise to adverse effects, including the accumulation of these substances in the thyroid and the development of conditions such as hypothyroidism and thyroid cancer. Compared to an iodine molecule, ROIs exhibit low affinity for conventional sorbents such as Ag@mordenite zeolite and triethylenediamine-impregnated activated carbon (TED@AC), resulting in lower sorption rates and capacities. Furthermore, in conditions resembling practical adsorption environments with high humidity, the presence of H₂O significantly impedes the adsorption process, leading to a nearly complete cessation of adsorption.

To address these issues, metal-organic frameworks (MOFs) can be effective alternative sorbents owing to their high surface area and designable and tailorable pore properties. In addition, the well-defined crystalline structures of MOFs render in-depth study on the structure-properties relationship. However, there has been limited research on the adsorption of ROIs using MOFs, with the majority of adsorption processes relying on highly reversible physisorption. This type of ROIs adsorption not only exists in a precarious state that is susceptible to volatilization but also exhibits significantly reduced adsorption capabilities in humid environments. Thus, for the secure adsorption of the volatile ROIs, the development of sorbents capable of chemisorption is highly desirable.

In this study, we focused on ROIs adsorption by electrophilic aromatic substitution with the electron-rich m-DOBDC⁴⁻ (m-DOBDC⁴⁻ = 4,6-dioxo-1,3-benzenedicarboxylate) present in Co₂(m-DOBDC). The chemisorption of ROIs via electrophilic aromatic substitution not only leads to the formation of C-C bonds, ensuring stability but also triggers color changes in the crystal by interacting with open-metal sites and iodide ions. Leveraging these advantages, we developed an infrared radiation-based sensing method that demonstrates superior performance, exhibiting high adsorption capacities and rates, even under the challenging conditions of high-humidity practical environments.

Keywords: Metal-organic framework, Radioactive organic iodides, Electrophilic aromatic substitution, Infrared radiation sensing

High Efficiency Adsorption of Selenite in Aqueous Solution Using Metal–Organic Frameworks

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Selenium (Se), a vital trace element found naturally, plays a pivotal role for human being in low concentrations. Notably, within the spectrum of essential elements, Se possesses the most restricted range between the dietary deficiency ($< 40 \mu\text{g day}^{-1}$) and the acute toxicity ($> 400 \mu\text{g day}^{-1}$). Therefore, it is of paramount importance to maintain bioavailable Se levels within permissible limits in our drinking water sources. Among the various Se species, inorganic variants such as selenite (SeO_3^{2-}) and selenate (SeO_4^{2-}) are highly water-soluble, with SeO_3^{2-} being notably more toxic than SeO_4^{2-} . Consequently, the primary focus lies in effectively sequestering SeO_3^{2-} from aquatic environments.

Numerous methods have been investigated for SeO_3^{2-} adsorption, including the use of metal oxides and carbon-based materials. Especially, iron oxides have garnered extensive attention due to their water stability and environmentally friendly properties. Nevertheless, their limited surface area and insufficient adsorption sites impose constraints on their efficacy as materials for SeO_3^{2-} removal. Recently, metal–organic frameworks (MOFs), composed of metal centers bridged by organic linkers have increasingly focused as promising adsorbents for SeO_3^{2-} removal, offering significant advantages such as large surface areas, high porosities, and structural versatility. Furthermore, there is a growing interest in defective MOFs, where intentional defects are introduced into the MOF structure. This deliberate introduction of defects aims to enhance the adsorption capacity by increasing the number of available adsorption sites.

In this context, herein, we present the Fe-BTC (BTC = 1,3,5-benzenetricarboxylic acid) synthesized via a post-synthetic metal-ion metathesis (PSMM) approach, which is one of the defect engineering methods applied to metal sites. We employ the well-established MOF, HKUST-1, known for its substantial surface area, as the pristine MOF. While the pristine MOF has a crystalline phase, during the PSMM process, Fe-BTC is transformed into an amorphous phase by allowing the introduction of numerous metal defect sites. These introduced metal defect sites serve as Lewis acidic sites, enhancing the adsorption capability for selenite. Furthermore, despite its amorphous nature, Fe-BTC exhibits a substantial surface area and porosity comparable to that of the crystalline pristine MOF. Consequently, Fe-BTC, distinguished by its numerous adsorption sites and its high porosity, demonstrates a remarkable capacity for selenite adsorption.

Keywords: Metal–organic frameworks, Selenite, Adsorption, Defect engineering

Development of an Extraction Method for Chelating Agents in Radioactive Concrete Waste

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Chelating agents, such as EDTA, NTA, and citric acid, possess the capacity to establish complexes with radionuclides, potentially enhancing the migration of such radionuclides from the disposal sites. Hence, quantification of these chelating agents in radioactive wastes is required to ensure secure disposal protocols. The determination of chelating agents in radioactive wastes is mainly composed of two steps, e.g. extraction and detection. However, there are little information on the extraction of the chelators in various radioactive wastes. We endeavored to optimize the extraction conditions for citric acid (CA) found within concrete, a prevalent component in the context of dismantled waste materials. Given the inherent high solubility of CA in water, we applied an aliquot of deionized water to the concrete and conducted a one-hour ultrasonic leaching procedure to facilitate chelate extraction. Subsequently, following the preparation of the concrete leachate via vacuum filtration and centrifugation to yield a clarified solution, we quantified the concentration of CA within the solution using Ion Chromatography (IC). To enhance leaching efficiency, we examined the % recovery variation with respect to the pH of the leaching solution. The optimized extraction method will be applied to diverse chelating agents and radioactive waste categories.

Keywords: Chelating agents, Extraction, Ion chromatography

Electrochemical Insights Into the Potentiostatic Uranium Titration Method

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The potentiostatic titration method is one of the effective methods for determining the total uranium assay in high-concentration uranium samples. A notable approach is the Devies-Grey titration method, which was first reported in 1964. In the U sample treatment process of this method, the reduction of U(VI) by Fe(II) is initially a non-spontaneous reaction based on the reduction potentials of the two half-reactions. However, in a high-concentration phosphoric acid medium, the reduction potential of Fe(II) is enhanced, simultaneously increasing the reduction potential of U(VI). As a result, the redox reaction becomes spontaneous due to these dual effects. On the other hand, the reaction kinetics can elucidate why nitric acid does not oxidize U(IV) during the oxidation of Fe(II) to Fe(III). Furthermore, the role of Mo(VI)/Mo(V) as a redox enhancer, employed alongside nitric acid, can be comprehended through electrochemical means. Similarly, the function of V(IV) as an electrochemical enhancer, aiding the action of the Cr(VI) titrant, becomes understandable. Grasping the various phenomena that manifest during the titration process is imperative for refining existing titration methods and pioneering new ones.

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Keywords: Electrochemical, Potentiostatic titration, Uranium, Devies-grey

Development of Sequential Radiochemical Separation of Pu and Am Isotopes in Radioactive Waste Samples Using Anion Exchange Resin and TRU Resin

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This study presents a rapid and sequential radiochemical separation method for Pu and Am isotopes in radioactive waste samples from the nuclear power plant with anion exchange resin and TRU resin. After radionuclides were leached from the radioactive waste samples with concentrated HCl and HNO₃, the sample was allowed to evaporate to dryness after filtering the leaching solution with 0.45 micron filter. The Pu isotopes were separated in HNO₃ medium with anion exchange resin. For leaching solution passed through anion exchange column, the Am isotopes were separated with TRU resin. The purified Pu and Am isotopes were measured by alpha spectrometer, respectively, after micro-precipitation of neodymium. The sequential radiochemical separation of Pu and Am isotopes in radioactive waste samples using anion exchange resin and TRU resin was validated with ICP-MS system.

Keywords: Radiochemical separation method, Pu isotopes, Am isotopes, Alpha spectrometer, Radioactive waste samples, ICP-MS system

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