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부산항국제전시컨벤션센터 (BPEX)



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

주최/주관

한국방사성폐기물학회

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1분과

핵주기정책·규제 및 비확산(Oral)

Development of Procedure for Setting Cybersecurity Exercise Scenarios With Case Study

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According to the “Law on protection and response measures for nuclear facilities and radiation”, Nuclear Power Plant (NPP) licensees should conduct periodic exercises based on hypothetical cyber-attack scenarios, and there is a need to select significant and probable ones in a systematic manner. Since cyber-attacks are carried out intentionally, it is difficult to statistically specify the sequences, and it is not easy to systematically establish exercise scenarios because existing engineering safety facilities can be forcibly disabled.

To deal with the above situation, this paper suggests a procedure using the Probabilistic Safety Assessment (PSA) model to develop a cybersecurity exercise scenario. The process for creating cybersecurity exercise scenarios consists of (i) selecting cyber-attack-causing initiating events, (ii) identifying digital systems, (iii) assigning cyber-attack vectors to a digital system, (iv) determining and adding type for operator’s response, (v) modifying a baseline PSA model, and (vi) extracting top-ranked minimal cut sets, and (vii) selecting a representative scenario.

This procedure is described in detail through a case study, an expected cyber-attack scenario General Transient-Anticipated Transient Without Scram (GTRN-ATWS). It refers to an accident scenario for ATWS induced by GTRN. Since ATWS is targeted for cyber training in some NPPs, and GTRN is one of the most common accidents occurring in NPPs, GTRN-ATWS was chosen as an example. As for the cyber-attack vector, portable media and mobile devices were selected as examples based on expert judgment.

In this paper, only brief examples of GTRN-ATWS events have been presented, but future studies will be conducted on an analysis of all initiating events in which cyber-attacks can occur.

Keywords: Cyber-attack, Cybersecurity exercise, Probabilistic safety assessment

Optimization of ISOCS Parameters for Uranium Quantification

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The characterization of nuclear materials is crucial for global nuclear safeguards efforts, as these materials can potentially be used for illicit purposes. In this study, we evaluated the applicability and performance of the In-Situ Object Counting System (ISOCS) equipment for the characterization and quantification of uranium, including uranium pellets and radioactive wastes.

Our methodology involved using ISOCS to measure samples with different enrichments and total amounts of uranium, and to analyze the results in order to evaluate the ISOCS's effectiveness in accurately characterizing the various uranium samples. To this end, we compared the ISOCS results with those of the Multi-Group Analysis for Uranium (MGAU) system, which is currently used in the field of international safeguards. The results of this study showed that the ISOCS was sensitive enough to analyze small amounts of uranium pellet, with %differences ranging from -0.7% to 19%. However, when analyzing shielded nuclear materials like in concrete waste, the uncertainty was relatively high, with %differences ranging from 11% to 67%. On the other hand, the MGAU system was unable to analyze uranium for the same spectrum, indicating the superiority of the ISOCS in terms of usability. The ISOCS instrument was also found to be effective in analyzing uranium in various types of samples without the need of standard sources.

Overall, the findings of this study have important implications for the development of more effective safeguards strategies for the characterization of nuclear materials. The ISOCS instrument could be a reliable tool for analyzing nuclear materials, contributing to global safeguards efforts to reduce the risk of nuclear proliferation.

Keywords: Safeguards, Gamma spectroscopy, Nuclear material, ISOCS

Safeguards Proposal for Milling Facilities Based on Nuclear Isotopic Ratios in Mill Tailings

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Milling facilities, which belong to the front end of the nuclear fuel cycle, are essential steps for utilizing uranium in nuclear power generation. These milling facilities currently provide the International Atomic Energy Agency (IAEA) with the location and annual production capacity of the facility through the Additional Protocol (AP, INFCIRC/540) and grant IAEA inspectors on-site sampling authority to apply safeguards to the facility. However, since milling facilities process a large amount of nuclear material and the product uranium ore concentrate (UOC) is bulk material, the absence of accounting for the facility could pose a potential risk of nuclear proliferation. Therefore, this study proposes technical approach that can be utilized for safeguards in milling facilities.

Since the half-life of uranium isotopes is much longer than that of its daughter, they reach the secular equilibrium condition. However, after milling process, the fresh tailing showed the break of that secular equilibrium. As time goes on, they newly reach another secular equilibrium condition. Based on this observation, this study discussed the feasibility of the ratio method in safeguards purpose. The scenario applied in this study was 1% of uranium mill tailing. It was observed that the U-238/Th-234 and U-238/Pa-234m ratios in fresh milling tails varied as a function of time after discharging, particularly during the first one year. This change can be worked as a significant signature in terms of safeguards. In conclusion, the ratio method in mill tails could be applicable for safeguards of nuclear milling facility.

Keywords: NORM, Milling facility, Additional protocol, Environment sampling, Bateman equation, Safeguards

Tritium Safeguards: Estimation of TPBAR Tritium Production and Nuclear Material Accountancy of TEF

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Nuclear fusion energy is considered as a future energy source due to its higher power density and no emission of greenhouse gas. Therefore, various researches on nuclear fusion is being conducted. One of the key materials for the nuclear fusion research is tritium because the D-T reaction is the main reaction in the nuclear fusion system. However, that tritium can also be used for non-peaceful purposes such as hydrogen bombs. Therefore, it is necessary to establish the safeguards system for tritium. In that regards, this study analyzed the possibility of applying safeguards to tritium.

To achieve this objective, the tritium production capacity through the light water reactor was analyzed. Tritium Production Burnable Absorber Rod (TPBAR) was modeled through the MCNP code, and tritium production was analyzed. The TPBAR is composed of a cylindrical tube with a double coating of aluminum and zirconium, filled with a sintered lithium aluminate (LiAlO_2) pellet to prevent the release of tritium. Tritium is produced by the reaction of Li-6 in the TPBAR with neutrons, and it is extracted and stored at the Tritium Extraction Facility (TEF). As a result, the tritium production increased as the burnup and Li-6 mass increased.

In addition, when the tritium produced in this way was transferred to TEF and extracted through the process, the application of safeguards measures was analyzed. To this end, various safeguards measures were devised, such as setting an Material Balance Area (MBA) for TEF and analyzing Material Balance Period (MBP). As there is no designated Significant Quantity (SQ) for tritium, cases were classified based on the type and form of nuclear weapons to estimate the Sigma MUF (Material Unaccounted For) of the TEF. Finally, the comprehensive application of safeguards to tritium was discussed. This research is expected to contribute to the establishment of IAEA safeguards standards related to tritium by applying the findings to actual facilities.

Keywords: Safeguards, Tritium, Tritium production burnable absorber rod (TPBAR), Tritium extraction facility (TEF), Nuclear material accountancy (NMA)

Development Plan for Next Phase of SMR Configuration Management System

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The configuration management system was implemented on the basis of a document management system that secured stable understanding, scalability, document security, and convenience in small modular reactor. To reduce the cost and risk of errors, configuration management is implemented to maintain a balance between design requirements, physical configuration, facility configuration information. In the initial stages, configuration change review procedures was developed with the main purpose of change management such as classification system management, configuration control committee management, configuration change review preparation, configuration control committee operation, follow-up measures, current status and tracking management. The preparation of the configuration change review consisted of preparation, distribution approval, designation of reviewers, review, collection of review opinions, and preparation of resolution results. In the operation of the configuration control committee, it was conducted by designating review members, reviewing members, collecting operation, and approval them.

The next step is to supplement and develop the requirements of IEEE Std 828-2012, such as configuration management planning, configuration management control, configuration identification, configuration change control, configuration status monitoring, configuration audit, interface management, and release management. Through this, issue raising, action management, and baseline management will be implemented.

Keywords: Configuration management system, Change management, Tracking management, Requirements management, Design requirements, Physical configuration, Facility configuration information, Consistency

Development of Diversion Path Analysis Tool and Its Applying Example

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To evaluate the safeguards system or performance in a facility, it is crucial to analyze the diversion path for nuclear materials. However, diversion paths can range from the extremely simplified to the complicated depending on the level of knowledge and the specific person conducting the analysis. This study developed the diversion path analysis tools using an event tree and fault tree method to generating diversion paths systematically.

The essential components of the diversion path were reviewed, and a logical flow was developed for systematically creating the diversion path. An algorithm was created based on the facility design components and logical flow, as well as the initial information of the nuclear materials and material flows. The event tree and fault tree analysis tools were used to test the path generation algorithm. The usage and limitations of these two logic methods are discussed, and ideas to incorporate the logic algorithm into practical program tools are suggested. The tests were analyzed on a typical light water reactor as an example, including automatic generation of dedicated pathways, configuration of safeguards measures, and analyzing paths with strategies for avoiding safeguard systems. The results led to the development of a draft pathway analyzer program that can be applied to general nuclear systems.

The results of this study will be used to develop a program module that can systematically generate diversion paths using the event tree and fault tree method. It can help to guide and provide practical tools for implementing SBD.

Acknowledgements

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

Nuclear Nonproliferation in Support of Exporting Nuclear Power

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The global nuclear nonproliferation regime has developed over the past 50 years based on the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) with three pillars: disarmament, nonproliferation and peaceful use of nuclear energy. Due to climate change and energy security in recent years, nuclear energy has been in the spotlight as an electricity generation source, and many countries are paying attention to introducing nuclear power plants (NPP). Whereas exporters pursue profit by selling their NPP, international organisations and member states that seek nuclear nonproliferation are concerned with potential proliferation risks by expanding the nuclear power industry worldwide. Simultaneously, the member states' right to peaceful use of nuclear energy has to be guaranteed as specified in NPT Article IV. Accordingly, the trade of nuclear power between the member states taking full responsibility is desirable from the nonproliferation perspective. This paper investigates whether the countries capable of exporting their nuclear power have complied with the global nuclear nonproliferation regime, deriving the role and position that South Korea is faced with, accordingly, has to take. The dynamics of exporters' competitiveness are discussed, emphasising that compliance with the regime must be considered a qualification when exporting NPP.

The achievement that South Korea has attained, fulfilling its role and responsibility under the regime, is highlighted. Since South Korea has developed the nuclear power industry in cooperation with the United States under the NPT and the ROK-US Agreement for Peaceful Nuclear Cooperation, the status quo of the two countries in the nuclear nonproliferation and industrial landscape is discussed. Among the newcomers who have officially announced the plan to introduce NPP, Saudi Arabia is put in a crucial position to aggregate or alleviate nuclear nonproliferation. To this end, the rationale for the ROK-US cooperation is proposed, evaluating the value of nuclear nonproliferation in support of exporting nuclear power.

Keywords: Nuclear nonproliferation, Nuclear power plant, Export strategy, International regime

1분과

핵주기정책·규제 및 비확산(Poster)

Analysis of Radiological and Physical Protection Emergency Plan for Nuclear Safety and Security Interface

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Domestic nuclear power plants have developed radiological emergency plans based on the USNRC's NUREG-0654/FEMA-REP-Rev.1 report and the Korea Institute of Nuclear Safety's (KINS) research report on radiation emergency criteria for power reactors (KINS/RR-12).

NUREG-0654 is a US emergency planning guide for nuclear power plants and provides detailed technical requirements for the content of radiological emergency plans. The document classifies radiological emergencies into three levels: Alert, Site Area Emergency, and General Emergency, which correspond to the white, blue, and red emergency levels used in domestic nuclear power plants.

KINS/RR-12 is a technical guidance document published by the Korea Institute of Nuclear Safety in 2012, which divides radiological emergency criteria into criteria for pressurized water reactors (PWRs) and criteria for boiling water reactors (BWRs), and describes in detail the regulatory position and implementation of radiological emergency criteria for domestic PWRs and BWRs.

The physical protection-related radiation emergency criteria included in the radiological emergency plan are specified in the radiological emergency criteria guidelines.

There are two items each related to white and blue emergencies and one item related to red emergencies. Standard order of emergency plan lists the physical protection-related radiological emergency criteria for domestic PWRs and BWRs, which are identical according to the radiological emergency criteria guidelines.

To enhance the physical protection regulation, the legal and regulatory basis for target set identification and vital area identification need to be established by considering radiological and physical protection emergency plan.

Keywords: Radiological emergency plan, Physical protection emergency plan, Nuclear safety and security interface, US emergency plan

Advancement Strategies for Nuclear Fuel Cycle Facility Operations

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The domestic representative nuclear fuel cycle facilities are post-irradiation examination facility (PIEF) and Irradiated Examination Facility (IMEF) at KAERI. They have regularly operated since 1991 and 1993, respectively. Due to the long period of use, the facilities are ageing, and maintenance costs are increasing every year. The maintenance methods have mainly been breakdown maintenance (BM) and partially preventive maintenance (PM). They involve replacing components that have problems through periodic inspections by on-site inspectors. However, these methods are not only uncertain in terms of replacement cycles due to worker's deviation on the inspection results, but also make it difficult to respond accidents developed through failures on the critical equipment that confines radioactive material. Therefore, an advanced operation and maintenance studied in 2022 through all of nuclear facilities operated at KAERI. Advancement strategy in four categories (safety, sustainability, performance, innovativeness) was analyzed and their priorities according to a facility environment were determined so a roadmap for advanced operation and maintenance could be developed. The safety and sustainability are higher importance than the performance and innovativeness because facilities at KAERI has an emphasis on research and development rather than industrial production. Thus, strategy for advancement has focused even more on strengthening the safety and sustainability. To enhance safety, it has been identified that immediate improvement of aged structures, systems, and components (SSCs) through large-scale replacement is necessary, while consideration of implementing an ageing management program (AMP) in the medium to long term is also required. Facility sustainability requires strengthening operation expertise through training, education, and cultivation of specialized personnel for each system, and addressing outstanding regulatory issues such as approval of radiation environment report on the nuclear fuel processing facilities and improvement work according to fire hazard analysis.

One of the safety enhancement methods, AMP, is a new maintenance approach that has not been previously applied, so it had to be thoroughly examined. In this study, an analysis was conducted on the procedure and method for introducing an AMP. An AMP for nuclear fuel cycle facilities was developed by analyzing the AMP applied to the BR2 research reactor in Belgium and modifying it for application to nuclear fuel cycle facilities. The ageing management for BR2 has the objective to maintain safety, availability and cost efficiency and three-step process. The first step is the classification of SSCs into four classes to apply graded approach. Secondly, ageing risk is assessed to identify critical failure modes, their frequency and precursors. Final step involves defining measures to reduce the ageing risk to an acceptable level in order to integrate the physical and economic aspects of ageing into a strategy for inspection, repair, and replacement. Similar approach was applied to the nuclear fuel cycle facility. Firstly, the SSCs of nuclear fuel cycle facilities have been classified according to their safety and quality classifications, as well as whether they are part of the confinement boundary. The SSCs involved in the confinement boundary were given more weight in the classification process, even if they are not classified as safety-class. A risk index for ageing was introduced to determine which prevention and mitigation measure should be chosen. By multiplying the health index and the impact index, the ageing risk matrix provides a numerical score that represents guidance on the prevention and mitigation of ageing effect. The health index is determined by combining the likelihood of failure and engineering evaluation of the current condition of SSCs, whereas the impact index is calculated by taking into account the severity of consequences and the duration of downtime resulting from a failure. This ageing management has to be thoroughly reviewed and modified to suit each facility before being applied to nuclear fuel cycle facilities.

Keywords: Nuclear fuel cycle facility, Safety, Sustainability, Ageing management program, Ageing risk, Operation and maintenance

Facility Safeguardability Assessment for the Light Water Reactor Spent Fuel Dry Storage Facility

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Owing to the increase in saturation rate of the spent fuel storage pond in the Kori nuclear power plant, the interim spent fuel dry storage facility is scheduled to be constructed at the Kori site. To implement safeguards in the new dry storage facility effectively, the concept of “Safeguards-by-Design” (SBD) should be applied to reflect nuclear safeguard provisions in the earliest design stages.

Detailed design information pertaining to dry storage facilities has not been determined; however, the design information related to safeguards have been inferred using case studies and interviews with nuclear power plant operators worldwide. On the basis of the results of the case studies on spent fuel dry storage facilities for light water reactors, most countries apply the metal cask method in containment buildings considering safety. Furthermore, Korean operators are also considering the same method owing to tight licensing schedules and safety issues.

Using the Facility Safeguardability Assessment (FSA) methodology (one of the safeguard evaluation methodologies), the difference in design between the heavy water reactor spent fuel dry storage facility, an established IAEA safeguards approach reference nuclear facility, and the light water reactor spent fuel dry storage facility (the new nuclear facility) were analyzed.

Two major differences were noted as issues pertaining to potential safeguards. First, the difference in design and transport method in terms of the difference in size and weight of the spent nuclear fuel is important; light water reactor fuel is 20 times heavier than heavy water reactor that needs partial defect inspection in assemblies. Second, the difference in safeguard approach owing to the difference between the modular storage method in heavy water reactor and the container type storage method in light water reactor must be considered; movable storage cask renders the IAEA surveillance approach difficult.

The results of this study can be used to identify the safeguards requirements in advance, enabling the operator to design new dry storage facilities resulting in timely and cost-effective implementation.

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Keywords: Safeguardability, Dry storage facility, Safeguards by design

A Study on the Application of Export Control Procedures in the ROK-UAE Administrative Agreement

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An administrative agreement (AA) was signed between NSSC and UAE FANR in January 2023 under Article 5 of the ROK-UAE Nuclear Cooperation Agreement. The AA aims to enhance regulatory efficiency in safeguards and export control. This study reviewed the export control measures for the items subject to the agreement (ISA) and implementation procedures under ROK-UAE AA by comparing them with other countries cases.

First of all, the ROK-UAE AA distinguishes between ISA and the inventory management target items. Technology is divided into two categories, one requiring consent for retransfer and the other, considering the characteristics of technology that is free to be copied and deleted, and thus less useful for inventory management. Only the former is included in the annual report, which differs from the ROK-Canada or ROK-Japan NCA, which includes all technologies subject to the agreements in the annual report. When ROK notifies export information, it is mandatory to specify whether the technology requires consent for retransfer. Furthermore, some technologies should be controlled as strategic information, even if excluded from the annual report, so efforts to prevent confusion are required.

Secondly, the ROK-UAE AA covers all items in INFCIRC/254/rev.9/part1, unlike the ROK-U.S. and ROK-Canada NCA, which listed equipment subject to them. This is significant because it clarifies the criteria for regulation by increasing the consistency between the trigger list items in the domestic law and the ISA. However, the expanded ISA scope could result in some changes in export control procedures. For example, when importing nuclear material (NM) from the US, only uranium was controlled as ISA, and the packages were not considered. In contrast, when exporting fuel assemblies (FA) for UAE, both uranium and zirconium cladding should be treated as ISA. To this end, NEPS was improved to implement the features of the ROK-UAE AA.

Consideration of the criteria and methods for imposing obligations under the agreement is essential because this is the first case of Korea concluded AA under exporting NPP and as a supplier of FA. Generally, the obligations for NM are imposed by the country of origin, conversion, and enrichment countries. Canada and EU recognize the fuel fabrication process as a substantial transformation and impose customs origin where the process takes place. Hence, NM fabricated from Canadian equipment is also subject to the same obligations as NM of Canadian origin. From this perspective, it would be appropriate to ensure ROK acts as a supplier and controls when exporting domestically manufactured FA. Moreover, a proper national obligation code system will be required to specify Korea's control rights.

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Keywords: Export control, Nuclear cooperation agreement, Administrative agreement, Nuclear material, Safeguards obligation

The Role and Prospects of the Resident Inspectors in KINAC

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KINAC began dispatching the resident inspector in 2012 to strengthen on-site Wolsong nuclear power plants (NPPs) regulations. The dispatched resident inspector is a member of the regional office of the Nuclear Safety and Security Commission (NSSC) and is in charge of technical support, on-site regulation of safeguards, and physical protection for the Wolsong regional office of NSSC. As the number of nuclear facilities in the ROK increased, the resident inspectors began to be dispatched to other regional offices. The resident inspectors were assigned to Hanul in November 2015, Kori in March 2017, Hanbit in March 2015, Saeul in March 2022, and Wolsong in March 2023. Accordingly, this paper intends to reflect on the increasing role of resident inspectors and predict on-site regulatory work in the field of nuclear control.

The role of the resident inspectors is described in detail in the internal regulations of KINAC. Among the tasks in the common field is technical support at regional offices for the most critical areas of nuclear control implementation, and on-site verification of the matters requested by the director of each implementation division shall be carried out. Tasks in the field of safeguards include an on-site check of facility regulation review, implementation of national inspections, technical support for IAEA inspections, and information management. Among them, technical support work for Unannounced inspections should be the top priority. These days, in particular, the importance of reviewing the results of checking advanced information and containment and surveillance equipment by facility operators is emerging. Among the tasks performed by the resident inspectors, more than 80% of the functions related to physical protection account for. The resident inspectors check the status of the physical protection system by weekly/monthly/quarter, implement physical protection regulation review and inspection, conduct exercise evaluation, and perform technical support for special assessments. Recently, regulatory activities related to radioactive terrorism and the emergence of illegal drones have been strengthened. In the field of cybersecurity, where its role has recently been increasing, the resident inspectors are performing basic field regulation tasks. Similar to the area of physical protection, the resident inspectors check the cybersecurity system for weekly, monthly, and quarterly readiness, and on-site inspections of cybersecurity review and inspection technical support, exercise evaluation, and other requests are mainly performed.

The role of the resident inspectors is expected to expand further in the future due to the increase in terrorist risks at home and abroad and changes in the regulatory environment. However, there is a limit to performing an increasing number of tasks, with the human resources of the resident inspectors limited to one to two for each site. If the resident inspectors are dispatched for each field of safety measures, physical protection, and cybersecurity, they can perform their duties more efficiently, but problems may arise in the operation of our personnel. Therefore, the proper and precise allocation of work while maintaining the current system is an essential part. The roles and prospects of the resident inspectors analyzed in this paper can be used to deploy the headquarters and field regulation personnel and set the direction of work in the future.

Keywords: Safeguards, Resident inspectors, Physical protection, Cybersecurity

Analysis and Improvement of Errors in Accounting Reports From 2021 to 2022

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According to the ROK-IAEA Comprehensive Safeguards Agreement (CSA), the ROK submits inventory change reports (ICRs), physical inventory lists (PILs), and material balance reports (MBRs). Suppose inventory changes occur in each material balance area (MBA). In that case, the facility operators prepare ICRs monthly, conduct physical inventory taking (PIT) every 12 to 18 months, and submit PILs and MBRs to KINAC. KINAC reviews ICR presented by the facility operators, submits it to the IAEA, and reports it to the Nuclear Safety and Security Commission (NSSC). Various methods have been prepared and implemented to minimize errors in reviewing the accounting reports submitted by the facility operators. Accordingly, this paper analyzes the mistakes in the accounting reports that occurred over the past two years and proposed methods to improve them.

The basis for carrying out the accounting reports is stipulated mainly in the CSA and the Nuclear Safety Act. First, Article 63 of the CSA describes the rationale for submitting the accounting reports, and the details are described in detail in the subsidiary arrangement. Article 98 of the Nuclear Safety Act stipulates information related to accounting reports, and details are described in the regulations on reporting internationally regulated materials, etc., of the NSSC Notice No. 2017-84. Among the accounting reports submitted in 2021, a total of 36 errors were confirmed. There were ten errors related to inventory changes, followed by six errors in the material balance period (MBP) in the header information. There were four cases of spacing, weight mismatch, and overdue errors, and the rest were related to grammar errors. There were a total of 30 errors in the accounting reports identified in 2022. MBP errors of header information, which occurred the second most in 2021, was the highest with nine, followed by six inventory change errors and five weight mismatch and overdue errors, respectively. Compared to 2021, the total number of errors has decreased by about six, which is interpreted as the result of outreach activities through accounting reporting workshops and nuclear control education conducted by KINAC.

Accounting reporting is the most critical part of the Nuclear Material Accounting and Control (NMAC) system. Efforts to check errors in accounting reports and improve report quality through outreach activities could be confirmed by the statistics of the two years analyzed earlier. In the future, if the reporting program used by the facility operators is improved to minimize errors and manage the accounting reporting system through continuous maintenance work, the quality of the accounting reports will be upgraded to the next level.

Keywords: Safeguards, Accounting reports, ICR, PIL, MBR

A Cyber Security Enhancement Method for Non-direct CDAs

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Nuclear Safety and Security Commission (NSSC) and KINAC review a Cyber Security Plan (CSP) by 「ACT ON PHYSICAL PROTECTION AND RADIOLOGICAL EMERGENCY」. The CSP contains cyber security implementation plans for the licensee's nuclear power plant, and it shall meet the requirements of KINAC/RS-015, a regulatory standard. The KINAC/RS-015 provides more detailed information on the legal requirements, so if licensees implement cyber security under the approved CSP, they can meet the law.

To protect nuclear facilities from cyber-attacks, licensees should identify their essential digital assets, so-called "Critical Digital Assets" (CDAs). Then, they apply cyber security controls (countermeasures for cyber-attacks) on CDAs consisting of technical, operational, and management security controls. However, it is hard to apply cyber security controls on CDAs because of the large amounts of CDAs and security controls in contrast to the shortage of human resources. So, licensees in the USA developed a methodology to solve this problem and documented it by NEI 13-10, and US NRC endorsed this document. The main idea of this methodology is, by classifying CDAs according to their importance, applying small amounts of security controls on less important CDAs, so-called non-direct CDAs.

In the case of non-direct CDAs, only basic cyber security controls are applied, that is, baseline cyber security controls. The baseline cyber security controls are a minimum set of cyber security controls; they consist of control a) from control g) a total of 7 controls. Although non-direct CDAs are less critical than other CDAs (direct CDAs), they are still essential to protect them from cyber-attacks.

This paper aims to suggest a cyber security enhancement method for non-direct CDAs by analyzing the baseline cyber security controls. In this paper, baseline cyber security controls were analyzed respectively and relatively and then concluded how to apply small amounts of cyber security controls on non-direct CDAs rather than direct CDAs without scarifying cyber security.

Keywords: Cyber security control, Baseline security control, Non-direct CDA

A Suggestion of Regulatory Improvement in Discarding CDAs Possessing Sensitive Information

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Licensees are required to protect critical digital assets (CDAs) in nuclear facilities against cyber-attacks, up to and including design basis threat (DBT), according to 「ACT ON PHYSICAL PROTECTION AND RADIOLOGICAL EMERGENCY」. However, CDAs may be excluded from cyber security regulations at nuclear power plant decommissioning, and this may lead to severe consequences if the excluded CDAs contain sensitive information such as the number and location of nuclear fuels and information on security officers. In that case, that information could be leaked to the adversary without adequately removing the information before discarding the CDAs. It can be potentially abused to threaten nuclear facilities inducing radiological sabotage and nuclear material theft. So, controls of sensitive information are needed. This study aims to derive regulatory improvements related to discarding CDAs that have sensitive information by analyzing foreign cases such as IAEA and U.S. NRC.

The sensitive information in the IAEA guide is the following: (1) details of physical protection systems and any other security measures in place for nuclear material, other radioactive material, associated facilities, and activities; (2) information relating to the quantity and form of nuclear material or other radioactive material in use or storage; (3) information relating to the quantity and form of nuclear material or other radioactive material in transport; (4) details of computer systems; (5) contingency and response plans for nuclear security events; (6) personal information; (7) threat assessments and security alerting information; (8) details of sensitive technology; (9) details of vulnerabilities or weaknesses that relate to the above topics; (10) historical information on any of the above topics.

In the case of the U.S. NRC, they categorize sensitive information into three groups: (1) classified information, (2) safeguard information (SGI), (3) sensitive unclassified non-safeguards information (SUNSI). Classified information is information whose compromise would cause damage to national security or assist in manufacturing nuclear weapons. The SGI concerns the physical protection of operating power reactors, spent fuel shipments, strategic special nuclear material, or other radioactive material. Finally, SUNSI is generally not publicly available information such as personnel privacy, attorney-client privilege, and a confidential source.

IAEA recommends protecting the above sensitive information in accordance with NSS No.23-G (Security of Nuclear Information), and NRC protects classified information, SGI, and SUNSI under relative laws. In the case of ROK, if security control measures are enhanced CDAs that possess sensitive information, the risk of information leakage will be decreased when those CDAs are discarded.

Keywords: Sensitive information, SGI, Discard

Establishment and Utilization of Semantic Segmentation Datasets for Detecting and Monitoring Nuclear Activities and Facilities Using Satellite Images

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Satellite imagery is an effective supplementary material for detecting and verifying nuclear activities and is helpful in areas where access and information are limited, such as nuclear facilities. This study aims to build training data using high-resolution KOMPSAT-3/3A satellite images to detect and identify key objects related to nuclear activities and facilities using a semantic segmentation algorithm. First, objects of interest, such as buildings, roads, and small objects, were selected, and the primary dataset was built by extracting them from the AI dataset provided by AIHub. In addition, to reflect the features of the area of interest (e.g., Yongbyon, Pyongsan), satellite images of the area were acquired, augmented, and annotated to construct an additional dataset (approximately 150,000). Finally, we conducted three stages of quality inspection to improve the accuracy of the training data. The training dataset of this study can be applied to semantic segmentation algorithms (e.g., U-Net) to detect objects of interest related to nuclear activities and facilities. Furthermore, it can be used for pixel-based object-of-interest change detection based on semantic segmentation results for multi-temporal images.

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Keywords: Satellite imagery, Nuclear activities, Semantic segmentation datasets

Exploring Nuclear Future Signals Using Scientific Article Data

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The purpose of this study is to detect future signals and changes in nuclear-related research to apply safeguards by design to new nuclear facilities or to determine nuclear fuel cycle-related research and development (R&D) activities. First, a total of 2,029 scientific articles published between 2015 and 2022 in the journal of “Nuclear Engineering and Technology” by the Korean Nuclear Society were collected. The authors of the scientific article used their expertise and knowledge to select keywords that can properly represent the article. Therefore, in this study, the keywords of each scientific article were analyzed using the technique of text mining. We then calculated the “word frequency” and “term frequency-inverse document (TF-IDF)” values of the keywords. Consequently, significant words such as “reactor,” “nuclear,” and “fuel” were extracted, which were represented as word clouds. Furthermore, keywords extracted through text mining were quantitatively classified into weak or strong signals using a keyword emergence map (KEM). The KEM is a tool that can explore future signals because essential keywords have a high frequency of appearance, and newer keywords are more important than older keywords. The KEM results showed no keywords in the strong-signal area in the field of nuclear academia. However, keywords such as “deep learning,” “earthquake,” “zircaloy,” and “CFD” were confirmed to be distributed in the weak signal area. A weak signal indicates the most probable topic that could become a strong signal in the near future. The weak signal methodology can be applied to predict future nuclear scientific trends in the rapidly changing world. Based on the results of the study, changes in the subject of nuclear-related scientific articles over the past eight years and future signals were interpreted. The results confirmed that this method can be applied to safeguards measures of new nuclear facilities in the design stage and can be used to detect R&D activities related to the nuclear fuel cycle in advance.

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Keywords: Safeguards by design, Nuclear fuel cycle, Text mining, TF-IDF, Keyword emergence map

A Review on the Catch-All Controls

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The concept of catch-all controls is not new, having been developed in the US and EU over 30 years ago and legislated about 20 years ago. It is a system that controls what is not in an item based on the end-user and end-use rather than the strategic items defined by the existing multilateral export control regimes. Most importantly, the controls are based on end-user or end-use.

Catch-all controls are the best tool to solve the problems of traditional list based controls. Technological advances, emerging technologies, and globalization have made standard export controls insufficient. Catch-all controls are optimized to control non-compliant, under-specified, partial, and intangible technology transfers.

Catch-all controls have been adopted and implemented in the four multilateral export control regimes (MECR) and are also reflected in legislation in the United States and the European Union. The NSG has a catch-all provision in INFCIRC/254/Part2 Paragraph 5, the AG has a catch-all condition in the Guidelines, and the MTCR has a catch-all provision in the Guidelines Paragraph 7. Korea has also implemented catch-all controls reflected in the Foreign Trade Act of 2004.

However, there are still many challenges of catch-all controls. First, the problem with catch-all controls is that there are different cases in different countries for list controls, and there are too many cases in other countries to control what is not on the list. Second, there is very little international legal enforcement for failing or missing catch-all controls. Finally, information about these catch-all cases is not uniformly available.

For catch-all control, Korea currently consolidates the list of traders of concern including UN sanction list and denial list of MECR, provides a correlation table of significant items and countries requiring catch-all control for stakeholders, and organizes and operates a coordination committee for traders of concerns. In the future, Korea should strengthen the monitoring of other countries' catch-all control cases and participate in international cooperation meetings to disseminate Korea's best practices.

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Keywords: UN sanction, Multilateral export control regimes, Export controls, Catch-all controls

A Study on the Necessity of ISO 21001 for Education on Nuclear Nonproliferation and Control by Comparing of ISO 9001 and ISO 21001

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Both ISO 21001 and ISO 9001 are standards developed by the International Organization for Standardization (ISO) for quality management systems. However, while ISO 9001 focuses on the general requirements for Quality Management Systems (QMS), ISO 21001 is specifically designed for educational institutions.

ISO 9001 is a widely recognized standard for QMS applicable to almost industries, including manufacturing and services. It defines the requirements for establishing, implementing, maintaining, and continuing improvement of QMS to improve customer satisfaction by meeting customer requirements and improving overall performance.

Meanwhile, ISO 21001 focuses specifically on educational institutions and is designed to develop and improve the curriculum efficiency by meeting trainees needs. It provides a system in which educational institutions can build, implement, maintain, and continuously improve the Education Management System (EMS) for the purpose of improving the satisfaction of trainees and other stakeholders. ISO 21001 covers a wide range of educational organizations, including schools, universities, and education providers.

KINAC/INSA, the Center of Excellence in Korea, is an educational institution in the field of nuclear control. So It has been developed and operated various international and domestic curriculum. KINAC/INSA obtained ISO 9001 certification in November 2016 and has been certified so far. However, in the scope of ISO 9001 certification, curriculum development process is not included so KINAC/INSA needs to obtain additional ISO certification specialized in education to improve the education quality. That is why KINAC/INSA is developing the ISO 21001 system, and aims to acquire certification in November 2023.

This paper explains the necessity for educational institutions to obtain ISO 21001 in comparison with ISO 9001. It also introduces the process of developing ISO 21001 system of KINAC/INSA. By implementing EMS based on ISO 21001, KINAC/INSA can expect to improve the educational satisfaction of trainees and other stakeholders through effective curriculum development and educational operation.

Keywords: ISO 9001, ISO 21001, Education, Nuclear

3D Training Scenarios Development for Inspection of Nuclear Material in Transit Using Virtual Reality Technology

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A person who performs or plans to conduct a physical protection inspection as stipulated by the law, the act on physical protection and radiological emergency, should obtain an inspector's ID card certified and authorized by Nuclear Safety and Security Commission Order No.137 (referred to as Order 137). In addition, according to Order 137, KINAC has been operating some training courses for those with the inspector's ID card or intending to acquire it. Also, strenuous efforts have been put to incrementally elevate their inspection related expertise.

Since Republic of Korea has to import uranium enriched less than 20% in order to manufacture fuels of nuclear reactors in domestic and abroad, the physical protection for categorization III nuclear material in transit is significantly important along with an increase in transport. The expertise of inspectors should be constantly needed to strengthen as the increase in transport leads to an increase in inspection of nuclear material in transit.

We have suggested a special way to improve the inspector's capacities through Virtual Reality technology (VR). A 3-Dimensional virtual space was designed and developed using a 3-axis simulator and VR equipment for practical training. HP's Reverb G2 product, which was developed in collaboration with VALVE Corporation and MicroSoft, was used as VR equipment, and the 3-axis motion simulator was developed by M-line STUDIO corp. in Korea for the purpose of realizing virtual reality.

The training scenarios of transport inspection consist of three parts: preparation at the shipping point, transport in route including stops and handover at the receiving point. At the departure point, scenario of the transport preparation is composed with the contents of checking the transport-related documents which should be carried by shipper and/or carrier during transport and confirming who the shipper and/or carrier is. Second, scenario is designed for inspector to experience how carrier and/or shipper protect the nuclear material during transport or stops for rests or contingency and how they communicate with each other during transport. Lastly, scenario is developed focusing on key check items during handover of responsibilities to the facility operator at the destination.

Those training scenarios can be adopted to strengthen the capabilities of those with inspector's ID card of physical protection in accordance with Order 137 and to help new inspectors acquire inspection-related expertise. In addition, they can be used for domestic education to promote understanding of nuclear security, or may be used for education for people overseas for the purpose of export of nuclear facilities.

Keywords: Inspector's ID card, Nuclear material in transit, Inspection of physical protection, Virtual reality

A Study on Security for Back-End Life Cycle of Nuclear Facilities

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In Korea, additional regulatory requirements are increasing due to the full-scale decommissioning of nuclear power plants following the permanent shutdown of Kori Unit 1 and Wolsong Unit 1. Accordingly, it is necessary to preemptively expand the scope of physical protection regulations from design, construction, and operation stage to back-end nuclear fuel cycle such as cessation of operation and decommissioning.

According to Article 2, Paragraph 24 of the Nuclear Safety Act, the decommissioning of nuclear facilities is defined as all activities to exclude them from the application of the Nuclear Safety Act by permanently suspending the operation of nuclear facilities, demolishing the facilities and sites, or removing radioactive contamination. In other words, it refers to a series of technologies or activities to safely and efficiently dismantle nuclear power plant and remove radioactive contamination and restore them to their original state after permanently shut down of nuclear power plant.

Security changes during decommissioning and decontamination since removing fuel from the reactor alters the plant's safety status, some of the systems or components considered as vital equipment during plant operation will no longer be needed. The vital areas may be reduced as fewer buildings, equipment and systems need to be protected, which means access controls, surveillance and so on can be reduced.

And also, decommissioning will probably require more workers than operation would, although this might not be the case at all times. From a security point of view, this might require more personnel or additional access points. Changing operating require changed security measures, to ensure that the required security level will be maintained while at the same time work proceeds efficiently.

Once all of the fuel is removed from the plant, radiological release risk is much lower. The lower risk requires a lower level of security measures. Even during the removal of nuclear material and contaminated equipment from nuclear facilities, lower level of security measures should meet regulatory requirements based on a graded approach.

Therefore, this study intends to examine the responsibilities and obligations of regulatory authorities, regulator, and nuclear operators in terms of protection after permanent shutdown and decommissioning.

Keywords: Physical protection, Decommissioning, Sabotage, Security measures

Considerations for Export Control of Nuclear Material, Deuterium and Heavy Water Accumulations

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Under the Foreign Trade Act, an export license from the Nuclear Safety Commission is required to export items specified in Part 10 of Schedule 2 of the Public Notice of Exportation and Importation of Strategic Items (Trigger List Items). In the case of nuclear materials, deuterium, and heavy water, its cumulative amount determines whether it is trigger list item. An export license is required only if the cumulative amount exported to a single end-user country from January 1st to December 31st exceeds the regulation criteria. The reason for this cumulative control is to exclude small amounts of materials from the scope of control as they are considered less important in view of nuclear proliferation, but to prevent the possibility of acquiring large quantities of materials by importing small amounts several times.

As a result, export control of nuclear material, deuterium, and heavy water requires different considerations than other Trigger List Items. First, materials exported by different companies must be consolidated to manage the cumulative amount. Second, it is necessary to continuously follow up the actual export status. If the material is not exported after it was classified as ‘non-Trigger List Items’, it should not be included in the cumulative amount. Third, there may be a difference between the accumulated quantities aggregated at the time of the classification and the time of the actual export. The classification should be changed if an export of the classified material is postponed or another export of same materials occurs before the export of the classified material. Fourth, the classification result of these materials should not be reused. Generally, the classification result could be reused within the expiration date (2 years) but in the case of substances. However, the reuse of classification result for materials should be limited as the classification results could be change depending on the cumulative amount. In addition, the sharing of classification results between different entities should also be restricted.

The government approval procedures are required even for export of small amounts of nuclear materials which are less than the regulation criteria. The cumulative quantities of nuclear materials are systematically managed in the Nuclear Export & imPort control System (NEPS) through these procedures. NEPS is also linked to the custom clearance system of Korea Customs Service, which enables to track actual exports and the time of exports. However, cumulative quantities for the heavy water and deuterium are managed individually by classification reviewers. The annual export plans are received in advance from major entities which deal with the materials for nuclear uses, and the cumulative quantities for each application are managed manually.

The systematic management has not been required as there were a few cases of exporting small quantities. However, systematic management may be required in the future as overseas expansion attempts from various companies in the nuclear field has been increasing. In addition, further study is needed on the criteria and system for calculating the cumulative amount. The time of aggregate the cumulative amount should be clarified by considering the difference between the time of classification and actual export. It is required to devise an efficient way to follow up the actual export.

Keywords: Trigger list items, Export controls, Nuclear material, Deuterium and heavy water, Nuclear export & import control system (NEPS)

A Study on Standards for Nuclear Materials Subject to the Nuclear Cooperation Agreement

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A bilateral Nuclear Cooperation Agreement (NCA) should define what is subject to the agreement and when. Nuclear Materials (NM) are the subject of NCA with almost all countries, and the definition used in these agreements is borrowed from Article 20 of the IAEA Charter. The IAEA's definition of NM as consisting of special fissionable material and source material and describes the types of material each contains.

In order to control the export of NM under national laws and implement NCA, not only the types of NM but also quantitative criteria are required. This is because controlling small quantities of NM is impossible, unnecessary, and would create excessive administrative burdens. For this reason, the NSG guidelines establish a quantitative threshold of NM requiring control. Nevertheless, no quantitative thresholds have been agreed upon for NM subject to a NCA. Whether NM transferred is subject to the NCA is primarily a matter for the supplier states to determine. The supplier states make the decision based on quantitative criteria defined in their own export control laws.

ROK identifies NM that require export licenses by reflecting the same criteria as the NSG guidelines in Foreign Trade Laws and its Notifications. Less than 500 kg of Natural Uranium, 1,000 kg of Depleted Uranium, 1,000 kg of Thorium, and 50 effective grams of special fissionable materials do not require an export license and is therefore not subject to NCA. In the US, the quantitative threshold for requiring an export license is different from that of ROK. For example, special fissionable materials that are not Pu are required if the individual shipment exceed 1 effective gram or 100 effective grams per year.

The difference in the quantitative thresholds for NM between the two countries mean that the same item may be subject to NCA under US standards, but not under ROK's. For example, the export of 8 grams of highly enriched uranium (93%) contained in a neutron detector would not be subject to the NCA in ROK, but would be considered NM subject to a NCA and required a special license in the US.

Of course, in order to ensure the application of safeguards and physical protection to all NM transferred between the two countries, the agreement may not include a quantitative threshold for NM. However, the absence of such a threshold can lead to different conclusions by the two countries on the same item and make it challenging to control retransfers. The definition of quantitative standards will be necessary in the supplementary administrative arrangement for the practical control and management of NM subject to the NCA.

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Keywords: Export control, Nuclear cooperation agreement (NCA), Nuclear material, Quantitative standards

A Study on the Necessity and Procedures for Nuclear Export Control Pre-Consultation

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In the late 80s, non-proliferation of weapons of mass destruction, including nuclear, chemical, and biological, became an internationally important issue, and in order to keep pace with the international situation, Korea amended the Foreign Trade Act in 1992 to legislate the export control of strategic goods. In addition, Korea joined the NSG, one of the international export control regimes, in 1995, and nuclear power operators are required to obtain export licenses in accordance with the NSG export control guidelines.

In the nuclear export business, technical documents, equipment, materials, SW, etc. are exported to the importer, and the export items may include strategic items designated by the NSG, so operators must check whether they are strategic items and, if so, obtain an export license in accordance with foreign trade laws that reflect the NSG export control guidelines. In the case of processing and exporting goods or materials imported from another country, exporters must fulfill complex nuclear export control procedures, including obtaining the original supplier's consent for re-transfer. In recent years, the international situation on export control has been more sensitive than ever, including the Russo-Ukrainian war, the U.S.-China semiconductor supremacy dispute, and U.S. lawsuits against domestic companies related to original technologies. In the worst-case scenario, a company may not be able to fulfill a contract due to export control issues.

In order to facilitate the smooth export business of operators when exporting nuclear energy, 'Nuclear Export Control Pre-Consulting' is proposed to check compliance with export control requirements in advance and provide operators with customized export control implementation plans reflecting business characteristics. Through the pre-consultation, issues, requirements, and preparatory documents related to nuclear export control can be checked before the export business starts in earnest, i.e., when the decision to participate in the business is made, and the business can be supported to export smoothly by complying with the export control system.

The most important aspect of pre-consultation is that domestic nuclear exporters need to know and apply for the program. To this end, the program will be actively promoted online and offline, and support will be provided for easy application through NEPS. In addition, procedures and outcomes will be continuously refined to ensure that the program is a means of ensuring full compliance with international nuclear non-proliferation norms for nuclear exports.

Keywords: Export control, Nuclear suppliers group (NSG), Nuclear export & import control system (NEPS), Pre-Consulting

A Case Study of DPRK's Illicit Trade Based on a UN Panel of Experts Report

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This study examined the Democratic People's Republic of Korea (DPRK)'s illegal trade in UN-sanctioned items as revealed in the UN panel of experts report in order to estimate the types of illegal trade in nuclear items, one of the UN-sanctioned prohibited items, and to find efficient ways to block it. Also, The research revealed that DPRK secretly imports UN-sanctioned prohibited items without going through customs through maritime transshipment, conceals or disguises them through identity laundering by falsifying documents at customs clearance, and makes various attempts to escape the international community's surveillance, such as using a combination of methods such as Re-Flagged and Double-Flagged for identity laundering, and concealing them without operating the Automatic Identification System (AIS) at sea.

The DPRK's Illicit trade cases have been divided into two types of transactions: those that violate customs clearance procedures by providing false information to customs through disguise or concealment, and those that do not go through normal customs procedures, such as smuggling. To block customs violations, technical measures such as increasing the number of inspections of container ships or improving the accuracy of inspections are required, while to block smuggling, since it does not go through physical inspections, there are ways to monitor it through satellite images or strengthen border enforcement such as airport bays and land routes.

As a result, DPRK's nuclear items are designated as sanctioned items under UN resolutions, and it is assumed that DPRK and its networks will attempt to trade illegally through a combination of customs clearance violations and non-customs clearance violations, depending on the circumstances. Furthermore, since DPRK is subject to extensive sanctions from the international community, including the UN, in connection with its nuclear weapons program, illegal trade continues, and efforts should be made to block illegal trade through physical inspection at customs clearance.

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Keywords: DPRK, UN panel of experts report, UN-sanctioned prohibited items, Nuclear item, Illicit trade

A Study on the Application of Internal Compliance Programs to South Korea's Nuclear Export Control

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The Internal Compliance Program (ICP) is a framework for promoting compliance with laws and regulations and minimizing violations. It aims to prevent law breaches, by raising awareness of the compliance within the organization, which leads to enhance the credibility of the organization, and to prepare for audits. From the perspective of nuclear export control, ICP can be used to verify the company's credibility by following NSG Guidelines and is expected to contribute to preventing the vertical and horizontal proliferation of nuclear weapons in the international community. However, ICP system is not globally established, and the NSG does not provide official guidelines for ICP. Therefore, this study aims to analyze the "Good Practices for Internal Compliance Programs for Nuclear and Nuclear-Related Exports" provided by the Pacific Northwest National Laboratory to find ways to apply and activate ICPs for domestic exporters.

The form of ICP could vary depending on company's size and internal environments, but it should be organized as follows. First, an internal department should be established so as to implement the ICP, and an executive who has export control knowledge should be assigned as the Chief Export Control Officer (CECO). The CECO, establish and revise ICP operating procedures and manual, organize contact point to communicate internally and externally. Second, measures should be established minimize risks in the export process, including business development transaction screening, supply chain, research and development, human resource, and intangible technology transfer risks.

Third, internal control system should be established for export control compliance. The CECO should conduct regular assessments to ensure compliance and strengthen the organization's internal export compliance processes. Fourth, an export-related training program should be periodically conducted for employees. In addition, as soon as the CECO becomes aware of, CECO should review the matter, take corrective action, and report to the relevant national authorities, when a violation of domestic export control laws or suspicious circumstances are captured.

Nuclear export control plays an important role in ensuring nuclear nonproliferation. Republic of Korea has been implementing the ICP system for Dual-Use Items under the Foreign Trade Act, but not for Trigger List Items. Therefore, introduction of ICP for Trigger List Items is expected significantly contribute to nuclear nonproliferation. The subjects of ICP will be initially targeted to major nuclear enterprises, then gradually expanded to all nuclear enterprises. Further researches are needed to introduce on ICP for Trigger List Items.

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Keywords: Internal compliance program (ICP), Export control, Trigger list, Non-proliferation

A Study on Rules of Internal Obligation Swaps for Nuclear Materials

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For the export of nuclear materials (NM), the NSG guidelines require governmental assurance from the importing State that the NM will be used for peaceful purposes, safeguards and physical protection will be applied, and prior consent will be obtained for retransfer. By providing this assurance, the importing State (recipient) is responsible for fulfilling the obligations required by the exporting States (supplier). If the Nuclear Cooperation Agreement (NCA) has been concluded between the supplier and recipient, it may be replaced by implementing the procedures under the NCA. In the case of NM subject to this obligation, continuous management at the national level is required because prior consent from the supplier may be required for retransfer to a third party under the assurance or may be subject to annual reporting.

The obligation swaps are the exchange of obligations of NM without the physical movement of it. Since the physical movement of NM is costly and risky, its obligations are often exchanged for commercial reasons.

The basis for obligation swaps is the fungibility and equivalence of NM. The fungibility allows that the inventories of NM need not physically identify the particular NM originally obligated but identify an equivalent quantity of the same isotopic composition. In addition, under the principle of equivalence, even if NM loses its unique physical properties, it can be exchanged by another obligated or non-obligated NM. That is, the principles of equivalence and proportionality allow the comparison of quantities of uranium in different forms. Therefore, it is theoretically possible not only to exchange obligations between NM in same physical form, but also different physical forms of same composition (with the same enrichment), e.g., UO₂ powder and its pellets.

In U.S., it appears that there are obligation swaps of NM between different enrichment levels, but according to the NCA and its Administrative Arrangement between ROK and U.S., Canada and Australia, the principle of fungibility and equivalence shall not be used to reduce the quality of a quantity of NM. In other words, swaps between NM of different enrichment levels are not allowed under the NCA and AA. However, according to the Supplementary Arrangement between ROK and Canada, the replacement of NM by lower quality NM may only occur where the two States so decide following consultation.

The U.S., Canada, and Australia, which are major suppliers of NMs, allow internal obligation swaps within the U.S. and the EU through NCA. The NCA between ROK and these countries does not address whether internal swaps are possible. Since governmental assurance does not impose restrictions on swaps, it can be considered if necessary. Although there is no actual practice of obligation swaps in ROK, research will be necessary regarding the extent to which swaps in ROK should be allowed and the need for government approval or permission.

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Keywords: Export control, National cooperation agreement (NCA), Governmental assurance (GA), Safeguards obligation, Obligation swaps, Principle of fungibility, Principle of equivalence, Principle of proportionality

Research and Analysis of Legislation for Cloud Conversion of Nuclear Export and Import Control System

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About 83% of the information systems of administrative and public agencies are operated by agencies, and most of them are vulnerable to security due to the small scale of operation, insufficient facilities, and lack of dedicated personnel. To address these issues, the Ministry of the Interior and Safety announced in June 2021 that, as part of the “Second Basic Plan for E-Government,” all information systems of administrative and public agencies will be converted to a cloud-based integrated management operating environment by 2025 to provide stable public services. Accordingly, relevant laws and guidelines should be researched and analyzed to prepare for the cloud conversion of the Nuclear Export and Import Control System (NEPS) operated by the Export and Import Control Office of the Korea Institute of Nuclear Nonproliferation and Control (KINAC).

The Cloud Computing Act defines cloud computing, establishes a basic plan and implementation plan, provides support for promoting the adoption of cloud computing by state institutions, supports the construction of integrated information and communication facilities based on cloud computing technology, provides cost and technical support, and regulates cloud security certification, and applies the Personal Information Protection Act and the Act on Promotion of Information and Communication Network Utilization and Information Protection to protect personal information. The E-Government Act defines integrated standards and principles for information resources, support for the use of cloud computing services, classification standards for information resources, and integrated standards for calculating the size and capacity of information systems. The Notice on Standards for Using Cloud Computing Services and Securing Safety for Administrative Agencies and Public Institutions specifies the standards for using cloud computing services and measures to secure stability for administrative agencies, contracts for using cloud computing services, and ensuring continuity of cloud computing services. The Basic Guidelines on National Information Security stipulate the establishment and implementation of security measures, system security, user security, security management, information and communication network security separating internal network and internet network, and cloud computing security measures, and stipulate the NIS security review when introducing private cloud services.

In order to convert NEPS to the cloud computing services, network, and software design plans, transfer plans, and cloud operation plans will be established in compliance with the relevant laws and guidelines. And future research will include researching the system status of major public and private cloud service providers and analyzing their advantages and disadvantages.

Keywords: Nuclear export, Nuclear import, Cloud, Cloud computing service, NEPS

Optimization of Semi-Supervised Deep Learning Model for Effective and Efficient Surveillance

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Surveillance plays a crucial role in safeguards. Reviewing surveillance data requires a significant number of inspection manpower. As the number of surveillance cameras increases, the demand for such manpower is expected to grow even more. Recently, in the field of security, there has been a development of deep learning models that automatically detect abnormal events from video images, and their usage is expanding. In this study, we used an AutoEncoder-based semi-supervised learning model, which can detect unexpected abnormal events, to detect anomalies in the UCSDped2 dataset and in simulating safeguards-related event videos taken at Dry Mockup facility of KAERI. To improve the model performance, we transformed the video images into two parts: the appearance part, which are sequences of video image frames, and the motion part, which are the pixel value differences of consecutive video frames. In addition, we added memory module to the bottle neck of the AutoEncoder model, and skip connection to enhance the model performance. To evaluate the model performance, we proposed a new evaluation index, which is adequate to the video images of safeguards surveillance in addition to the widely used AUC (Area Under the ROC Curve).

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Keywords: Surveillance, Deep learning, Semi-supervised learning, Autoencoder

Factors Affecting Information Quality: Considerations on Analyzing Nuclear Proliferation Activity

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When proliferation activity occurs, states and non-state actors combine various sources of information to gain a better understanding of the situation. The quality of information source, content, and presentation can significantly influence the perception of decision-makers and end users. However, a state's nuclear or missile activities are almost always classified. Also, states might intentionally reveal information to deter their adversary, threaten their adversary, bolster their prestige in the international community, or a combination of all three. Hence, any revealed information inherently contains some degree of uncertainty regarding its credibility. Lack of credibility makes it difficult for other states or non-state actors to determine how much of the information is accurate and how much is deception. The increase of publicly available information (open-source information) and the development of tools to collect, process, and analyze this information increased the possibilities of using open-source information to cross-check the proliferation claims of states. North Korea is a hermit state that has very little outside interaction. It also has continued to develop and refine its nuclear program. How credible is information released by North Korea? How can scholars/experts compensate for this lack of credibility in information on North Korea's nuclear activity? This paper seeks to apply a framework on information quality to answer these questions. First, it will briefly explain the factors comprising information quality (sources, content, and presentation). Then, it will apply the information quality framework to North Korean activity analysis. It will conclude with implications of using the information quality framework to analyze a state that is low in accessibility and high uncertainty.

Keywords: Information quality, North korea, Nuclear activity, Nonproliferation, Credibility

A Study on the Analysis of Accident Scenarios at Spent Fuel Storage Facilities and Evaluation of the Protective Performance of Spent Fuel Storage Facilities Against Design Basis Threat

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In the wake of the Fukushima NPP accident, research on the safety evaluation of spent fuel storage facilities for natural disasters such as earthquakes and tsunamis has been continuously conducted, but research on the protection integrity of spent fuel storage facilities is insufficient in terms of physical protection.

In this study, accident scenarios that may occur structurally and thermally for spent fuel storage facilities were investigated and safety assessment cases for such scenarios were analyzed. Major domestic and international institutions and research institutes such as IAEA, NEA, and NRC provide 13 accident scenario types for Spent Fuel Pool, including loss-of-coolant accidents, aircraft collisions, fires, earthquakes. And 10 accident scenario types for Dry Storage Cask System, including transportation cask drop accidents, aircraft collisions, earthquakes.

In the case of Spent Fuel Pool, the impact of the cooling function loss accident scenario was mainly evaluated through empirical experiments, and simulations were performed on the dropping of spent nuclear fuel assembly using simulation codes such as ABAQUS. For Dry Storage Cask System, accident scenarios involving structural behavior, such as degradation and fracture, and experimental and structural accident analyses were performed for storage cask drop and aircraft collision accidents. To evaluate the safety of storage container drop accidents, an empirical test on the container was conducted and the simulation was conducted using the limited element analysis software.

Among the accident scenarios for spent fuel storage facilities, aircraft and missile collisions, fires, and explosions are representative accidents that can be caused by malicious external threats. In terms of physical protection, it is necessary to analyze various accident scenarios that may occur due to malicious external threats. Additionally, through the analysis of design basis threats and the protection level of nuclear facilities, it is necessary to derive the probability of aircraft and missile collision and the threat success probability of fire and explosion, and to perform protection integrity evaluation studies, such as for the walls and structures, for spent fuel storage facilities considering safety evaluation methods when a terrorist attack occurs with the derived probability.

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Keywords: Terrorism, Sabotage, Spent fuel storage, Physical protection, Design basis threat

Analyzing the Results of Nuclear Export and Import Control System (NEPS) Functionality Improvement

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The Nuclear Safety and Security Commission (NSSC) and the Korea Institute of Nuclear Nonproliferation and Control (KINAC) operate the Nuclear Export and Import Control System (NEPS), an online comprehensive export and import control management system to guide the domestic nuclear export and import control regime and efficiently process the application, processing, and issuance of various civil complaints required for compliance. This paper analyzes the results of NEPS functional improvements made in 2022.

First, NSSC and KINAC launched a nuclear plant technology follow-up system. Since establishing the Nuclear Plant Technology Export License (Plant License) system in 2015, large-scale nuclear power plant projects with a large amount of technology transfer have been issued a Plant License, allowing them to transfer technology without a separate individual export license. The recipient of a Plant License is not required to obtain an additional export license but to follow up on the transferred technology, such as checking whether it is a strategic technology and reporting quarterly transfer details. A dedicated system has been established to facilitate the follow-up of plant license projects. That has improved work efficiency for both the regulated and the regulator.

Second, we have improved the procedures for retrieving and supplementing civil petitions. We added a function allowing civil petitioners to directly retrieve a civil petition to cancel a previously filed civil petition or to revise it themselves. In addition, we improved the procedure for supplementing a civil petition by setting a supplementation deadline for a reviewer's request to modify a civil petition and sending a reminder before the expiration of the supplementation deadline. That enhances the convenience of the complainant and the examiner.

Finally, we have changed the numbering system for export and import control reviews. Previously, the application number was a 15-digit system with an 8-digit date and 7-digit serial number. We have simplified the number by changing the three unused digits in the serial number to a single symbol that separates the type of application. That made it possible to utilize the number for relevant searches in the future. Also, simplifying the number has improved the efficiency of applying, processing, and consulting for civil affairs.

In the future, NSSC and KINAC will continue to upgrade NEPS and make various functional improvements, and there will be many changes in system operation through cloud transfer in a few years.

Keywords: Nuclear export, Nuclear import, NEPS, Functional improvement, System upgrade

A Study on the Effectiveness of Introducing a Plagiarism Detection System as a Research Performance Management Tool in Nuclear Regulatory Authority

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With the introduction and implementation of the National Research and Development Innovation Act in 2021, researchers are required to have a greater understanding of research ethics and to comply more strictly. The range of misconduct in research and the standards for sanctions have been expanded with the introduction of the National Research and Development Innovation Act. In addition, research-performing institutions and specialized agencies have been obligated to establish their own research management systems and standards according to the changed criteria. The Korea Institute of Nuclear Nonproliferation and Control (KINAC), a nuclear regulatory authority that is conducting national R&D in related fields, has sought to strengthen research ethics by revising related regulations, introducing a plagiarism detection system, and expanding related education in accordance with these policies.

In this study, we analyzed the effectiveness of the plagiarism detection system as a basic quality control measure for research results and a tool for enhancing research ethics, which was introduced. KINAC did not simply introduce a plagiarism detection program but established institutional improvements and other regulatory measures to support it, with the aim of more effectively managing research results. To analyze the effectiveness of this system, we calculated the plagiarism rate by sampling 30 papers each year for the three years before the introduction of the plagiarism detection system. When comparing the plagiarism rates before and after the introduction of the plagiarism detection system, no exceptional cases of high plagiarism rates were found in papers published after the introduction of the system. Although most of the papers before the introduction of the system showed a satisfactory plagiarism rate, some cases showed high plagiarism rates. We analyzed the cause of such cases in detail. Some exceptional cases were also found to be included in the range of misconduct regulated by the National Research and Development Innovation Act. As no such cases were found after the introduction of the system, we could infer that the system is effectively functioning as a tool for basic quality control and enhancing research ethics.

In the future, we plan to expand the sample qualitatively and quantitatively by including other forms of outcomes published by the institution, not just papers, and conduct a more detailed analysis. Based on the results, we will develop various improvement plans for enhancing the quality and research ethics of the institution's research results.

Keywords: Research performance, Plagiarism detection system, Nuclear regulatory authority

A Study on the Quality Management Standards and Application for Improving R&D Activities as a Regulatory Support Process

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Since the National R&D Innovation Act was enacted in 2022, it became a crucial issue how to qualify or improve R&D activities and disseminate their outcomes. Many organizations have referred to various quality management standards such as the American National Standards Institute/American Society for Quality (ANSI/ASQ) Z1.13, International Organization for Standardization (ISO) 9001, and the American Society of Mechanical Engineers Nuclear Quality Assurance-1 (ASME NQA-1), as a means to set up their own quality system. ISO is the international standard for implementing a quality management system (QMS), which provides a framework and principles for managing an organization's QMS, with the aim of ensuring that the organization consistently provide products or services that meet regulatory requirements. ISO 9001 can cover all aspects of an organization's operations, and it can also be expanded to include R&D areas. The introduction of ISO 9001 to R&D aims to improve R&D practices and establish a standardized process framework for conducting R&D. ANSI/ASQ Z1.13 provides quality guidelines for research and consists of 10 sections covering various aspects of research quality, emphasizing ethical conduct, clear objectives, reliable data collection, and analysis. ASME NQA-1 is one of quality assurance standards for nuclear facility applications, but it has been extended and applied to R&D activities in the nuclear fields. It just focuses on planning, procedures, documentation, competence, equipment, and material control. KINAC has conducted extensive research on verifying and regulating nuclear activities while providing support for national nonproliferation technologies and policies. In addition to the quantitative growth achieved so far, efforts are being made to establish a qualitative and integrated management system. As a first step to achieve this goal, this study reviewed international standards and methodologies for research quality and derived the key components for R&D quality management. Moreover, the appropriate outline of quality management system framework was proposed for R&D as a regulatory support process, based on the ISO 9001. The implementation of quality management standards and procedures for R&D in KINAC, which could lead to improved research practices, more reliable data collection and analysis and increased efficiency in conducting R&D activities.

Keywords: Quality management, Regulatory R&D, Regulatory support process, International standards

A Case Study of the Implementation of ISO/IEC 17025 for Quality Management System in Testing Laboratory

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The Korea Laboratory Accreditation Scheme (KOLAS) is the national accreditation body responsible for providing accreditation services to testing and calibration laboratories. The primary objective of KOLAS is to promote the quality and reliability of laboratory testing by providing nationally and internationally recognized accreditation services. Laboratories accredited by KOLAS are required to meet rigorous international standards set by the International Organization for Standardization (ISO) and are subject to regular assessments to ensure ongoing compliance with the standards. KOLAS accreditation is highly regarded both domestically and internationally, and is recognized for providing high-quality and reliable testing services.

The nuclear analysis laboratory at KINAC has been working to establish a quality management system to ensure the external reliability of analytical results and to secure its position as an authorized testing agency. To achieve this, a detailed manual and procedure for nuclear material analysis were developed to conform to the international standards of ISO/IEC 17025. This study presents the preparation process for establishing the management system, focusing on meeting technical and quality requirements for the implementation of the ISO/IEC 17025 standard in the KINAC nuclear analysis laboratory, specifically in the field of chemical testing (dosimetry, radioactive, and neutron measurement subcategories). The preparation process involved two tracks. The first track focused on satisfying technical requirements, with Thermal Ionization Mass Spectrometer (TIMS) and Inductively Coupled Plasma-Mass Spectrometer (ICP-MS) selected as the major equipment for analysis. Analytical methods for determining isotope ratios and concentrations of nuclear materials were determined, and technical qualification was ensured through participation in proficiency test programs, inter-experimenter comparison tests, and uncertainty reports. The second track focused on developing the quality system, including quality manuals, procedures, and guidelines based on the requirements of the ISO/IEC 17025 standard. Various implementation documents were produced during the six-month pilot period, in accordance with the three levels of documents required by the standard. Implementation of ISO/IEC 17025 is expected to have a systematic quality management process for the analysis lab's operations and to increase confidence in KINAC's nuclear analysis.

Keywords: Nuclear material analysis, ISO/IEC 17025:2015, KOLAS, Quality management

Performance Evaluation for Commercialization of Real-Time Seal Device

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Timely detection of nuclear activity is important for the management and supervision of nuclear materials, and inspections on North Korea as a method of safety measures for this monitoring system seem to be a not far future in the rapidly changing North Korea relationship.

However, a simpler and more reliable monitoring device is required since the inspection period is limited and the possibility of revisiting is unclear. The seal is a simple but easily used security device for monitoring forgery and falsification in the IAEA.

IAEA presents matters related to 1) project engineering, 2) remote monitoring, and 3) seal development as the three major tasks of the Department of safeguards and Division of Technical Support. The importance of development, modernization, and application of new seal devices is emphasized, and advanced sealing and identification system development is in progress at many research institutes such as JRC, ISCN, and JAEA.

Since the existing seal devices used by IAEA and KINAC can only be confirmed through on-site inspections for damage, it is difficult to respond immediately in the event of similar situations such as theft of nuclear materials and loss of continuity of knowledge. Unlike facilities that comply with the requirements for safety measures, such as domestic nuclear facilities, in the case of facilities subject to denuclearization, it is very likely that various hazardous environments will exist that make it difficult to apply safety measures.

Hence, a real-time seal device has developed through prior research due to the high possibility of situations in which Continuity of Knowledge (COK) is not maintained, such as damage, malfunction, and power loss of sealing and monitoring equipment.

Through previous studies, the real-time seal device was loaded with server-based operating software and improved its performance by utilizing feedback from real users (KINAC) after use.

In this study, the effectiveness of the previously developed sealing system was verified through performance evaluation, and the authentication of the equipment was secured through environmental tests.

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Keywords: Real-time, Seal, Active, Remote

Neutron Source Strength Evaluation Based on Spent Nuclear Fuel Characteristics

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Nuclear inspection is necessary to verify nuclear activities. If North Korea takes denuclearization, North Korea's nuclear materials should be verified through non-destructive testing and destructive testing for nuclear material production. Since destructive testing of all substances is impossible, non-destructive testing is essential. Most non-destructive tests are performed by measuring the energy of gamma rays, but the characteristics of nuclear fuel can be evaluated by measuring neutron sources when enclosed with thick shields and when shielding structures are difficult to remove. Before the neutron source evaluation of MAGNOX used by North Korea, the relative characteristics will be evaluated later by analyzing the burnup, enrichment, and cooling time of the spent nuclear fuels discharged from the domestic nuclear power plant. This study evaluated the source strength and major nuclides according to burnup for the WH17×17 nuclear fuel assembly.

The depletion calculation was conducted using SCALE 6.2 ORIGEN, and 3.5wt% enrichment, 10, 20, 30, 40, 50, 60 MWd/kg burnup, and five years cooling time, the minimum requirement for transport specified in the notice of the Nuclear Safety Commission, was applied. Although the impact assessment on enrichment should be evaluated with MCNP Tally to consider the fission reaction of the generated neutrons, this study only evaluated the spontaneous fission and (a, n) reactions that occurred first because it only evaluates the burnup impact.

As burnup increased, neutron generation increased, and most of the total neutron strength occurred through spontaneous fission from the 10 MWd/kg burnup step. The influence of Pu-240 nuclides was dominant in the 10 MWd/kg burnup step but most neutrons were generated in tiny amounts of Cm-244 generated from 20 MWd/kg burnup. Since DPRK's 5 MWe Yongbyon MAGNOX has very low burnup (about 0.7 MWd/kg), the primary neutron sources of 10 MWd/kg, Am-241 and Pu isotopes, especially Pu-240, are expected to be used as indicators for evaluating spent nuclear fuel characteristics. If only specific nuclides are evaluated as major neutron sources at lower burnup than those evaluated in this study, in that case, the accuracy of non-destructive testing can be improved. Additionally, the evaluation according to the enrichment and cooling time should be considered as well.

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Keywords: Denuclearization, Spent nuclear fuel, Neutron strength, Burnup, Enrichment, Cooling time

Nuclear Posture Strategy of the DPRK and the Republic of Korea's Response

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The DPRK had been upgrading its nuclear weapons capabilities from the past to the sixth nuclear test in 2017, and Kim Jong-un has been in power since the death of Kim Jong-il in 2011, striving to upgrade and diversify four nuclear tests and firing means. In 2022, in particular, DPRK launched more than 40 ballistic and cruise missiles and provoked them in various ways, such as developing solid rocket engines, flying fighter jets, and invading drones. In addition, reprocessing facility activities have been observed again since 2021 at the Yongbyon Nuclear Research Complex. Operational activities such as continuous activities of the 5MWe Yongbyon reactor and the additional construction of new buildings are observed. DPRK's recent activities could result in nuclear weapons in all except conventional weapons provocations.

DPRK has researched and developed nuclear weapons since the 1950s. It has been preparing to operate nuclear weapons, operating nuclear power, and modernizing nuclear power simultaneously. Given the number of nuclear weapons using DPRK's nuclear materials and various means of missiles, an offensive transition is expected to enable restrictive deterrence strategies that can be used first use and on assured retaliation. In addition, based on the nuclear strategy, which is interpreted as Vipin Narang's nuclear doctrine and nuclear posture, DPRK is also capable of assured retaliation and asymmetric escalation posture. In particular, considering the continuous activities of the Yongbyon Nuclear Research Complex, which has recently diversified the investment vehicle, and the delegation of nuclear weapon use, it is expected to move differently from the previous one based on the changed nuclear strategy. However, there are clear limitations to interpreting it as a completely assured retaliation and asymmetric expansion nuclear strategy. First, there is a lack of development of atmospheric re-entry vehicles that can avoid ICBM interception for assured retaliation capabilities against the United States. Second, there are limitations in the open capacity of nuclear weapons due to the absence of SSBN capabilities. However, delegation to operations at strategy force suggests the possibility of asymmetric expansion strategies.

The previous analysis of DPRK's nuclear strategy and limitations is valid in that the U.S. nuclear umbrella guarantees the Republic of Korea in a strong alliance between the Republic of Korea and the U.S. If the Republic of Korea lacks a nuclear umbrella due to the weakening of the alliance or limits U.S. intervention by having more than dozens of ICBMs, it is considered that DPRK can use a definite confirmation retaliation and asymmetric escalation nuclear strategy. As a response to this, it is the first way to verify and strengthen the validity of the three-axis system (Kill Chain, KAMD, KMPR), second to strengthen the Korea-U.S. alliance, and finally to appeal to the international community and increase consensus. In particular, it is possible to form a consensus of sanctions and condemnation DPRK by expressing concerns about nuclear dominoes caused by nuclear proliferation and arms competition to the international community.

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Keywords: DPRK, Nuclear weapon, Nuclear posture, Nuclear strategy

Algorithm Optimization of Inspection Sample Size Calculation for Improving Computation Time

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A sample size calculation algorithm was developed in a prototype version to select inspection samples in domestic bulk handling facilities. This algorithm determines sample sizes of three verification methods satisfying target detection probability for defected items corresponding to one significant quantity (8 kg of plutonium, 75 kg of uranium 235). In addition, instead of using the approximation equation-based algorithm presented in IAEA report, the sample size calculation algorithm based on hypergeometric density function capable of calculating an accurate non-detection probability is adopted. The algorithm based the exact equation evaluates non-detection probability more accurately than the existing algorithm based on the approximation equation, but there is a disadvantage that computation time is considerably longer than the existing algorithm due to the large amount of computational process. It is required to determine sample size within a few hours using laptop-level performance because sample size is generally calculated with an inspector's portable laptop during inspection activity. Therefore, it is necessary to improve the calculation speed of the algorithm based on the exact equation.

In this study, algorithm optimization was conducted to improve computation time. In order to determine optimal sample size, the initial sample size is calculated first, and the next step is to perform an iterative process by changing the sample size to find optimal result. Most of the computation time occurs in sample size optimization process performing iterative computation. First, a non-detection probability calculation algorithm according to the sample sizes of three verification methods was improved in the iterative calculation process for optimizing sample size. A computation time for each step within the algorithm was reviewed in detail, and improvement approaches were derived and applied to some areas that have major effects. In addition, the number of iterative process to find the optimal sample size was greatly reduced by applying the algorithm based on the bisection method. This method finds optimal value using a large interval at the beginning step and reduces the interval size whenever the number of repetitions increases, so the number of iterative process is less than the existing algorithm using unit interval size. Finally, the sample sizes were calculated for 219 example cases presented by the IAEA report to compare computation time. The existing algorithm took about 15 hours, but the improved algorithm took only about 41 minutes using high performance workstation (about 22 times faster). It also took 87 minutes for calculating the cases using a regular laptop.

The improved algorithm through this study is expected to be able to apply the sample size determination process, which was performed based on the approximate equation due to the complexity and speed issues of the past calculation process, based on the accurate equation.

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Keywords: Sample size calculation, Inspection sampling plan, Bulk handling facility, IAEA inspection, Non-detection probability, Algorithm optimization

Improvement of Cyber Security Contingency Plan by Classifying CDA Types

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The licensee of nuclear facilities in the Republic of Korea should ensure the functionality of Critical Digital Assets (CDAs) is maintained and minimize the negative impact of cyber-attacks by establishing a cyber security contingency plan. The contingency plan should include detailed response guidelines for each stage of detection, analysis, isolation, eradication, and recovery and comply with the requirements specified in KINAC's "Regulatory Standard 015 - Security for Computer and Information System of Nuclear Facilities". However, since the cyber security contingency plan describes the overall response guidelines for CDA, it may be difficult to respond practically to cyber-attacks.

This paper suggests a method to address this issue by performing exercises based on the classification of CDA types. CDAs in nuclear facilities can be classified according to their characteristics. The criteria for classifying CDA types include whether the asset is a PC, whether communication ports (RS-232, 422, 485) exist, whether storage devices can be connected through USB/memory card ports and whether internal settings can be changed through HMI devices such as built-in buttons. By classifying CDA types based on the proposed criteria, the attack vectors of CDAs can be defined. By defining the attack vectors, a list of cyber-attacks that CDAs may face can be created, and abnormal symptoms of CDAs resulting from the listed cyber-attacks can be defined. By using the defined abnormal symptoms of CDAs, the response measures of detection, analysis, isolation, eradication, and recovery can be concretized and reflected in the contingency plan. This may enable a more practical emergency response.

This paper presents an improvement to the cyber security emergency response plan through the definition of cyber-attacks based on the classification of CDA types. By improving the contingency plan for CDAs as a whole using the proposed method, it is expected that more effective response measures can be taken in the event of a cyber-attack.

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Keywords: Cyber security, Cyber-attack, Critical digital asset, Contingency plan

A Study on Nuclear Material Production Estimation Using Probabilistic Methodology

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The Korea Institute of Nuclear Nonproliferation and Control (KINAC) is developing a simulation model to estimate nuclear material production. This model is a foundational technology in interpretation and evaluation in preparation for denuclearization verification. Through this model, it is possible to estimate the amount of nuclear material that can be produced based on information on the activities of facilities related to the nuclear fuel cycle in the actual denuclearization verification stage. This model makes it possible to determine whether the declared amount of nuclear material is reliable.

In addition, the reliability of the reported information can be confirmed through on-site inspection. However, there is a possibility that proliferation-related activities cannot be detected even through this inspection, and a normal state may be misdiagnosed as carrying out nuclear proliferation-related activities. Therefore, it is unreasonable to specify activities related to nuclear proliferation with only one inspection. Since each inspection method has its diagnosis rate and false diagnosis rate, measures such as repeating the same inspection method or combining different inspection methods are required to detect activities related to nuclear proliferation reliably.

Therefore, a model capable of estimating the number of repetitions to obtain a reliable nuclear activity detection probability was developed by using each inspection method's diagnosis rate and false diagnosis rate as input information through a Bayesian inference method. Through this model, it can be concluded that repetitive inspections increase the probability of detecting nuclear proliferation-related activities.

This approach confirmed the possibility of repeatedly breaking away from the high-intensity inspection method that causes political and diplomatic resistance from the target country and substituting it with a more readily acceptable, low-intensity inspection method.

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Keywords: Nuclear non-proliferation, Bayesian inference, Probabilistic methodology

Current Status of Technology for Extracting Uranium From Seawater

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Given the limited terrestrial reserves of uranium (about 4.6 million tons), exploring alternative resources is essential to ensure the long-term supply and sustainability of nuclear energy. Uranium extraction from seawater (UES) is a potential solution to this issue since the amount of uranium dissolved in seawater (about 4.5 billion tons) is approximately 1000 times that of terrestrial reserves. However, the ultra-low concentration of uranium in seawater (about 3.3 ppb) makes it a challenging task to make UES economically feasible.

This paper provides an overview of the current status of UES technology, which has evolved over the past seven decades. Starting from inorganic adsorbents such as hydrous titanium oxide in the 1960s, amidoxime-based fibrous adsorbents gained the most attention until the early 2010s due to their ease of deployment in actual seawater conditions and high affinity for uranium. Nowadays, research on organic adsorbents with microstructures is prevailing due to their ability to easily control surface area and compositions. In addition, this study identifies the key issues that need to be addressed to make UES technology economically viable.

Keywords: Uranium, Seawater, Extraction, Adsorbent

An Analysis of Canadian Revised Cybersecurity Regulatory Guide for SMRs

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Recently, more than 70 SMRs have been developed around the world due to their modularity, flexibility, and miniaturization. An innovative SMR (i-SMR) is also being developed in Korea, and operators are planning to apply for a Standard Design Approval (SDA) in 2026 after completing the standard design. Accordingly, regulatory organizations are conducting R&D on regulatory requirements and guidelines for systematic SMR standard design review by referring to IAEA and NRC cases.

In terms of security, SMRs are expected to undergo many changes not only in terms of physical security through security systems, security areas, and vital equipments, but also in terms of cybersecurity through new digital technologies, remote monitoring, and automated operation. Accordingly, the IAEA Fundamental Safety Principles (SF-1) require operators to improve the safety of nuclear facilities by considering security requirements, access control requirements, and the results of operational impact assessments based on threats from the design and construction stages. Similarly, the U.S. nuclear regulatory body (NRC) has confirmed the status of security assessment and design considering design basis threats (DBTs) in the NuScale standard design review process, and the Canadian nuclear regulatory body (CNSC) has revised security regulatory guidelines and applied them to the SMR standard design review.

Among these various activities related to SMR security, this paper analyzes the major changes in the cybersecurity regulatory guidelines for SMRs recently revised by the CNSC, the Canadian nuclear regulatory body. Compared to the previous guidelines, the Defensive Cybersecurity Architecture (DCSA), including external logical access control, security level and zone communication requirements, verification and validation (V&V) activities during development phases, and system & service acquisition security requirements have been added. Other changes, such as the cyber incident response program, will be analyzed and compared. Through the revised regulatory guidelines, the CNSC has divided cybersecurity levels into four (High, Moderate, Low, and Business), strictly prohibiting remote access to High and Moderate levels, and allowing remote access to Low levels only for maintenance purposes. In addition, the paper will analyze the detailed revisions, such as prohibiting access to the High level from lower levels and allowing only handshaking signals from the Low level to the Moderate level.

Keywords: Small modular reactor, Canadian nuclear safety commission, Cyber security, Regulatory guide

Analysis of Threats to Nuclear Facilities Using PoisonTap Report

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On April 28, 2022, a North Korean hacker (operator) recruited an active officer in exchange for virtual currency to steal military secrets and attempted to hack the battlefield network, and it was revealed that he tried to use PoisonTap during the investigation. Let's analyze whether these events pose a threat to nuclear facilities.

PoisonTap is a tool coded in the Node.js language to Raspberry Pi Zero and weaponized. When connected to a target PC with a USB or Thunderbolt port, the target PC can be occupied in about a minute. PoisonTap materials include Raspberry Pi Zero, a USB expansion port that can be connected to the target PC, a space where the code can operate (code about 12 MB), and Node.js (weaponizable code), which can be made without much difficulty. PoisonTap's functionality allows cookies and sessions to be stolen through hijacking and allows remote access by exposing internal routers to the outside. Some of the reasons why PoisonTap occurred are that network devices connect directly to the computer without any conditions. And one of the big problems of this vulnerability is the design problem of the Internet itself, so it is difficult to block or defend technically. It is difficult to protect if it is simply a software problem because it is different from how to fix it through software code modification.

According to the PoisonTap principle analysis, it connects the PoisonTap to the target PC based on the network's characteristics (subnets of lower-priority network devices are given higher priority than gateways of higher-priority network devices). The HTML+Javascript generated while being connected becomes a backdoor that can be connected anytime. In other words, by creating a Websocket that can be connected to the web browser itself at any time, an attacker can connect to the target PC at any time.

In such a threat, PoisonTap is used to break in and install a web backdoor on the target PC to make it continuously accessible and attack even if the PoisonTap is disconnected. This problem is believed to be an insider threat not only to military units but also to nuclear facilities that are closed networks. PoisonTap can be brought into major nuclear areas in cooperation with insiders with general maintenance of USB equipment. Ordinary workers often leave their laptops or leave them for a while by inserting a screen-saver password. In addition, because there is no communication with the outside, actions that do not seal USB ports and enter deep sleep mode (network connection) can be exposed as cyber threats to nuclear facilities using PoisonTap by malicious insiders.

Keywords: Poisonsnap, Cyber security, Threat, Network

Application Methods of Process Monitoring to the Nuclides Management Process to Enhance Safeguardability

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As the use of nuclear energy has been expanded, issues in a spent nuclear fuel management are raised. Several methods have been proposed and developed to manage spent fuels safely and efficiently. One method is to reduce environmental burden in disposal of spent fuels by decreasing volume of high-level waste. A nuclides management process (NMP) is one example. Through this novel process, it is able to separate highly mobile nuclides (ex. iodine, krypton), high thermal emission nuclides (ex. strontium, barium), and optionally, uranium from spent fuels.

Since the NMP is a back-end fuel cycle technology, a reliable safeguards system should be employed in the facility. As international atomic energy agency (IAEA) recommends safeguards-by-design (SBD), it is desirable to investigate an appropriate safeguards approach at a step of technology development. Process monitoring (PM) is a complementary safeguards technology for traditional safeguards technologies which based on mass balance. PM traces nuclear materials indirectly but consecutively by using process parameters such as temperature, pressure, and flow of fluid. These parameters are obtainable by installing appropriate sensors. In a respect of SBD, PM is a promising approach to achieve the safeguards goal, the timely detection of diversion of a nuclear material. However, it is necessary to classify useful process parameters from all available signals which provided from PM in order to properly utilize PM.

In this study, we investigated application methods of the PM approach to NMP. NMP consists of several unit processes in series. Firstly, we inspected a principle and a feature of each unit process. Based on the results, we evaluated applicability of the PM approach to each unit process according to effectiveness in enhancing safeguardability. Several unit processes were expected that their safeguards are able to be enhanced by using certain process parameters from PM.

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Keywords: Spent fuel treatment, Nuclides management process, Nuclear safeguards, Process monitoring

2분과

사용후핵연료 처분전관리(Oral)

Cold Spray Deposition Coatings to Extend Lifetime of Spent Nuclear Fuel Storage Stainless Steel Canisters

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Long-term safe storage of spent nuclear fuel (SNF) determines sustainability of the current light water reactor (LWR) fleet. In the U.S., SNF is stored in stainless steel canister in dry cask storage system (DCSS) after spending several years in wet pool storage system while there is no DSCC in Republic of Korea. The SNF storage time in DSCC is expected to be multiple decades since no permanent geological repositories are identified in both countries. One limiting factor for extended storage of SNF in DSCC is chloride-induced stress corrosion cracking (CISCC) in the welded regions of the stainless steel canisters. The propensity for the occurrence of CISCC has warranted the development of the mitigation and repair technologies to ensure the safe and long-term storage for both present and new canister although no CISCC failure was reported yet.

This study investigates cold spray deposition coatings of 304 L and 316 L stainless steels on prototypical stainless steel canisters such as sensitized flat and C-ring samples. The cold spray technology has been identified as the most promising approach by Extended Storage Collaboration Program (ESCP) driven by Electric Power Research Institute (EPRI). The talk includes microstructural characterization, adhesion strength measurement, residual stress evaluation, and corrosion behavior of the coated materials in boiling MgCl_2 solution and electrochemical corrosion tests in NaCl solution. In addition, the capability of repair of cracks on the canister surface using the coating technology will be presented.

Keywords: Spent nuclear fuel, Dry cask storage canister, Cold spray technology, Chloride-induced stress corrosion cracking

Investigation on the Safety of Boeun U Ore Using Multiple Environmental Tracers

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Uranium inventory in Boeun aquifer is facing the artificial reservoir that intended for supplying water to nearby cities (40-70 m apart) where, toxic radionuclides might mobile and enter the reservoir. In order to understand U mobility in the system, groundwater and fracture filling materials (FFMs) were analyzed for microbial signatures, C, O, Fe, S and U-series isotopes. The $\delta^{18}\text{O-H}_2\text{O}$ and ^{14}C signatures suggested groundwater was originated from upland recharges dominantly and not affected by mixing with the surface water. However, the $^{234}\text{U}/^{238}\text{U}$ activity ratios (ARs) and $^{230}\text{Th}/^{234}\text{U}$ ARs in FFMs ranged from 0.93 to 1.67 and from 0.22 to 1.97, respectively, indicating that U was mobile along the fractures. In shallow FFMs, the U accumulations (~ 157 mg/kg) were found with Fe enrichments (~ 226798 mg/kg) and anomalies of $\delta^{56}\text{Fe}$ and $\delta^{57}\text{Fe}$, implied U mobility in shallow depths was associated with Fe-rich environment. Also, in the shallow depths, Fe-oxidizers, Gallionella was prevailing in groundwater while Acidovorax was abundant near U ore depth. The Fe-rich environment in shallow depths was formed by pyrite dissolution, demonstrated using $\delta^{34}\text{S-SO}_4$ and $\delta^{18}\text{O-SO}_4$ distribution. Conclusively, the Fe-rich aquifer was capable of immobilizing the dissolved U through biotic and abiotic processes, without significant discharge into the nearby reservoir.

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Keywords: Uranium, Natural analogue, U-series disequilibria, Geological disposal, Age-dating, Safety case

Effect of Mechanical Properties Under Dynamic Loading Condition of Austenitic Stainless Steel for Spent Fuel Storage

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For Korean nuclear fuel cycle project, it is necessary to design and evaluate the integrity of spent fuel storage. For the design and evaluation of spent fuel storage, it is necessary to evaluate the properties of various materials used in spent fuel storage. The materials previously considered in the design of nuclear power plants were limited to static properties and were listed in design and manufacturing code and standards. However, for the evaluation of the storage containers in scenarios such as transportation and events, dynamic material property evaluations are required. Research on the dynamic properties of materials is generally conducted in the fields of automotive and aerospace, and most of the studies are on metal materials under sheet conditions. Since the structural materials of the storage containers for used nuclear fuel are mostly composed of thick materials, consideration should be given to property evaluation methodology and quantitative comparison. In this study, the mechanical properties of stainless steel material with canister application were evaluated according to the strain rate, and the crack resistance evaluation was also performed. It was confirmed the changes in strength and crack resistance according to the increase in strain rate and observed differences in microstructural hardening behavior.

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Keywords: Dynamic loading condition, Strain rate, Austenitic steel, Spent fuel storage

Pressure Limit Analysis Code for Spent Nuclear Fuel Transport Cask Submerged in Deep Sea

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In the event of a loss of a SNF (spent nuclear fuel) transport cask during maritime transportation, it is essential to evaluate the critical depth at which the integrity of the cask can be maintained under high water pressure. SNF transport casks are classified as Type B containers and the integrity of the containment boundary must be maintained up to a depth of 200 meters unless the containment boundary was breached under beyond-design basis accidents. However, if an intact SNF cask is lost at a depth deeper than 200-meter, release of radioactive material may occur due to breach of containment boundary with over-pressure.

In this study, we developed a code for the evaluation of the pressure limit of SNF transport cask, which can be evaluated by inputting the main dimensions and loading conditions of cask. The evaluation model was coded as a computer module for ease of use. In the previous study, models with three different fidelities were developed to ensure the reliability of the calculation and maintain sufficient flexibility to deal with various input conditions. Those three models consisted of a high-fidelity model that provided the most realistic response, a low-fidelity model with parameterized simplified geometry, and a mathematical model based on the shell theory. The maximum stress evaluation of the three models confirmed that the mathematical model provides the most conservative results than the other two models. The previous results demonstrate that mathematical models can be used in the code of computer modules.

In this study, additional models of transport cask were created using parametric modeling techniques to improve the accuracy of the pressure limit assessment code for different cask and situations. The same boundary conditions and loading conditions were imposed as in the previous simplified model, and the maximum stress results considering the change in the shape of the transport container were derived and compared with the mathematical model. The comparison results showed that the mathematical model had more conservative values than the simplified model even under various input conditions.

Accordingly, we applied the mathematical model to develop a transportation container pressure limit evaluation code that can be simulated in various situations such as shape change and various situations.

Keywords: SNF transport cask, Integrity, Parametric modeling, Evaluation code

Drying Experiments Using a Lab-scale Vacuum Drying Apparatus

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There is a need to develop a quantitative residual water measurement method to reduce the measurement uncertainty of the amount of residual water inside the canister after the end of vacuum drying. Therefore, a lab-scale vacuum drying apparatus was fabricated and its characteristics were evaluated by performing vacuum drying experiments based on the amount of residual water, vacuum drying experiments based on the surface area of residual water, and vacuum drying experiments based on the energy of residual water using the lab-scale vacuum drying apparatus. As a result of the vacuum drying experiments, if the surface area of water is the same, the greater the amount of water, the greater the energy of the water, so more energy is transferred to the surface of the water. Therefore, more water evaporated, and the average temperature of the remaining water was higher. The larger the surface area of the water, the more energy it takes to vaporize it, so the faster it dries and the faster the drying time. Before ice formed, energy was actively transferred by conduction heat transfer from the top, center, and bottom of the water to provide the energy needed for the water to evaporate from the surface. However, no energy was transferred from the water just before it turned into ice. When vacuum drying water, you can dry more water if you dry it slowly over a longer period of time. Therefore, by using a vacuum pump with a low flow rate, the pressure can be lowered slowly to prevent ice from freezing, thereby improving the drying quantity. It was evaluated that there was a good agreement between the energy used when water evaporated and the energy absorbed from the surroundings to within about 4%. Therefore, if the energy absorbed from the surroundings is known, it is possible to evaluate the amount of water evaporated in vacuum drying.

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Keywords: Vacuum drying, Residual water, Canister, Lab-scale, Conduction

Introduction of International Seismic Tests for SNF Dry Storage Casks

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Currently, in the United States, Spent Nuclear Fuel (SNF) is stored at the Independent Spent Fuel Storage Installations (ISFSIs) at 73 Nuclear Power Plants (NPPs). The SNF inventory stored on-site either in pools or dry storage was 84,500 MTU in 2020. The inventory stored in on-site dry storage facilities was 39,207 MTU (46% of the total), and it is growing at a rate of approximately 3,500 MTUs per year. However, because a site for geologic repository for permanent disposal of SNF has not been constructed in the U.S., the SNF will need to be stored in dry storage facilities across the U.S. for a much longer period of time than originally planned. During this time, the dry storage facilities could experience earthquakes of a different magnitude than the one for which they were originally designed.

However, there is little data on the response of SNF inside dry storage systems to seismic loads in the U.S., and the various gaps and nonlinearities between storage containers, canisters, baskets, aggregates, and fuel make it very difficult to evaluate by analytical methods. Therefore, a full-scale shake table test is being planned as an international joint research project led by Sandia National Laboratories (SNL) in the U.S.

In Korea, KNF decided to participate in this seismic test through the project of SNF integrity evaluation under road and sea normal transportation conditions organized by KNF and conducted by KORAD, KAERI, and Kyung-Hee University, and has provided the KNF 17ACE7 and PLUS7 test assemblies for the tests to SNL.

The test will be conducted at the LHPOST6 shake table test facility operated by University of California in San Diego (UCSD) from 2023 to 2024, with the participation of KNF, CRI, and KAERI in Korea. The test units consist of a NUHOMS 32 PTH2 canister, a mockup of a generic vertical cask, a mockup of a generic horizontal storage module, 4 surrogate fuel assemblies, and 28 dummy assemblies.

The seismic inputs for the tests will consist of ground motions (acceleration time histories) representative of hard rock, soft rock, and soil sites and seismic conditions in moderately tectonically active Central and Eastern US and highly tectonically active Western US. Ground accelerations for soft rock and soil conditions will be developed taking in account soil-structure interaction.

Not only is this test almost impossible to conduct independently in Korea in terms of scale, facilities and costs, but it is also considered an essential test for those of us who are preparing for dry storage of spent nuclear fuel, given the increasing social concern about earthquakes due to the recent earthquake in Turkey.

Keywords: Seismic, Spent nuclear fuel, Dry storage, Storage cask, Ground motion

Emulation of Radiation Damage in Al-B₄C Neutron Absorber Used in Temporary/Interim Spent Nuclear Fuel Storage Facility

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Al-B₄C neutron absorbers are currently widely used to maintain the subcriticality of both wet and dry storage facilities of spent nuclear fuel (SNF), thus long-term and high-temperature material integrity of the absorbers has to be guaranteed for the expected operation periods of those facilities. Surface corrosion solely has been the main issue for the absorber performance and safety; however, the possibility of irradiation-assisted degradation has been recently suggested from an investigation on Al-B₄C surveillance coupons used in a Korean spent nuclear fuel pool (SNF). Larger radiation damage than expectation was speculated to be induced from $^{10}\text{B}(n, \alpha)^7\text{Li}$ reactions, which emit about a MeV α -particles and Li ions. In this study, we experimentally emulated the radiation damage accumulated in an Al-B₄C neutron absorber utilizing heavy-ion accelerator. The absorber specimens were irradiated with He ions at various estimated system temperatures for a model SNF storage facility (room temperature, 150, 270, and 400°C). Through the in-situ heated ion irradiation, three exponentially increasing level of radiation damages (0.01, 0.1, and 1 dpa or displacement per atom) were achieved to compare differential gas bubble formation at near surface of the absorber, which could cause premature absorber corrosion and subsequent ^{10}B loss in an SNF storage system. An extremely high radiation damage (10 dpa), which is unlikely achievable during a dry storage period, was also emulated through high temperature irradiation (350°C) to further test the radiation resistance of the absorber, conservatively. The irradiated specimens were characterized using HR-TEM and the average size and number density of radiation-induced He bubbles were measured from the obtained bright field (BF) TEM micrographs. Measured helium bubble sizes tend to increase with increasing system (or irradiation) temperature while decrease in their number density. Helium bubbles were found from even the lowest radiation damage specimens (0.01 dpa). Bubble coalescence was significant at grain boundaries and the irradiated specimen morphology was particularly similar with the bubble morphology observed at the interface between aluminum alloy matrix and B₄C particle of the surveillance coupons. These characterized irradiated specimens will be used for the corrosion test with high-temperature humid gas to further study the irradiation-assisted degradation mechanism of the absorber in dry SNF storage system.

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Keywords: Al-B₄C neutron absorber, Spent nuclear fuel (SNF), Temporary/interim storage facility, Radiation damage, Heavy-ion irradiation, Irradiation-assisted degradation

Reduction of Zirconium Oxide Compounds for Pyroprocessing Application of Corium

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Pyroprocessing is a promising technique for the treatment of damaged fuel debris (corium) generated by severe nuclear accidents. The debris typically consists of (U, Zr)O₂ originating from the UO₂ fuel and Zr alloy-based cladding. By converting the corium to a metallic form, the principal components of the fuel can be recovered through subsequent electrorefining, allowing for long-term storage or final disposal. A study investigated the reduction of zirconium oxide compounds by Li metal as a reductant in molten LiCl salt. This research explored the feasibility of treating damaged nuclear fuel debris, which mainly consists of (U, Zr)O₂. The results showed that ZrO₂ was successfully reduced to Zr metal by Li metal in LiCl salt at 650°C without the formation of Li₂ZrO₃. In particular, Zr metal was produced without the formation of Li₂ZrO₃ when LiCl salt containing a high concentration of Li metal was used. However, Zr metal was produced with Li₂ZrO₃ when LiCl salt containing both Li metal and Li₂O was added. This suggests that the concentration of Li metal in the LiCl salt is an important factor in determining the formation of Li₂ZrO₃. The study also demonstrated that Li₂ZrO₃ was partially reduced to Zr metal by Li metal in LiCl salt. This finding suggests that Li metal may be effective in reducing other oxide compounds in molten LiCl salt, which could be useful in the treatment of corium. Overall, the research provides valuable insights into the feasibility of using pyroprocessing for the treatment of corium. The ability to recover and store the principal components of the fuel through electrorefining could have important implications for the long-term management of nuclear waste.

Keywords: Lithium metal reduction, Molten LiCl salt, Zirconium compound, Corium, Pyroprocessing

Indigenous SF Mechanical Integrity Evaluation Scheme (K-Method) Under Normal and Accident Drop Failure Modes During Transportation Reflecting Characteristics of Dry Storage Degradation

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Integrity evaluation scheme for Spent Fuel (SF) dry storage has been developed under transportation failure modes. This method especially considered the degradation characteristics of Spent Fuel (SF) during dry storage such as radial and circumferential hydride content, hydride volume fraction, oxide thickness, etc. Hydride and zircaloy cladding are considered as material composite system, using correlation models related to material properties. Critical Strain Energy Density (CSED) is compared with Strain Energy Density (SED), to evaluate cladding integrity. CSED serves as material characteristics, while SED can be considered as boundary condition. To calculate the CSED of cladding in the lateral failure mode, circumferential hydride concentration is used. SED is calculated considering both the bending moment and axial load. On the other hand, in the longitudinal failure case, fuel rod temperature, internal pressure, hoop stress, radial hydride concentration is used to calculate CSED. And pinch force (contact) was considered to evaluate SED. Model validations were conducted by comparing hot cell SF test and existing validated evaluation results. To separately handle normal transportation conditions from hypothetical accident conditions, SED according to stress-strain analysis results was separated into elastic and plastic regions. As a result of applying this scheme for 14×14 SF, failure probability of normal condition was zero, which is the similar result with DOE and same with EPRI. Regarding accident condition, lateral case showed similar result, but longitudinal case showed different but reasonable result, which was due to the different analysis conditions. The proposed methodology which was indigenously developed through this study is named as K-method.

Acknowledgements

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Keywords: Spent nuclear fuel, High burnup, Hydride, Reliability assessment, Mechanical integrity

A Study on Zr-fire Phenomenon During Loss-of-Coolant-Accident in a Spent Fuel Pool

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When a loss of coolant accident which causes a partial or a full drainage in the SFP would happen, Zircaloy-4 spent fuel cladding begin to react with high temperature air, and the heat generates by exothermic reaction between Zircaloy-4 cladding and surrounding air. Due to the heat, the ignition may occur in the surface of Zircaloy-4 cladding. If the Zr-fire phenomenon occurs during the accident in a SFP, the spent fuel cladding and pellets would be severely fragmented and powdered and it may bring about a massive release of radioactive source terms. Therefore, it is crucial to prevent the zirconium fire phenomenon for the spent fuel pool safety. However, a main cause to trigger the zirconium fire was not identified. In order to identify a possible mechanism of the Zr-fire phenomenon, OECD-NEA SFP Project I, II was initiated. In this paper, we reviewed the Zr-fire phenomenon which may occur in the spent fuel pool for complete loss of coolant accident scenario.

The Spent Fuel Pool Project (hereinafter SFP project) is the experimental program to investigate the phenomena of spent fuel pool complete loss of coolant accident using a 17×17 PWR fuel assembly. In this section, the zirconium fire phenomenon which was observed from the SFP project is briefly investigated. This paper presented the fuel assembly temperature (i.e. zirconium alloy cladding temperature) and oxygen concentration profile of the SFP project phase-1 ignition test. At around 12.7 hour, the temperature abruptly increased and the oxygen concentration also dramatically decreased. This abrupt temperature escalation is the zirconium fire phenomenon. In order to investigate the mechanism of this zirconium fire phenomenon, behaviors of both temperature and oxygen concentration were fully compared.

This paper reviewed the results of OECD-NEA SFP project experiment and then a mechanism of Zr-fire phenomenon was discussed. It seems that the Zr-fire phenomenon might be a consequence of thermal mismatch between heat generation and dissipation. A large amount of heat might be generated by the air oxidation of Zircaloy-4 spent cladding immediately after the kinetic transition which is a breakaway phenomenon. This paper discussed the relationship between the breakaway phenomenon and the Zr-fire phenomenon in case of air oxidation of Zircaloy-4 spent cladding. This paper presents preliminary findings on the Zr-fire phenomenon from the open experiment data of the prototypic spent fuel severe accident scenario. These findings would enhance the understanding of Zircaloy-4 spent cladding air oxidation and severe accident scenario progression in a SFP.

Keywords: Spent fuel pool, Zircaloy-4 spent fuel cladding, Zr-fire phenomenon, Loss-of-coolant-accident, OECD-NEA SFP project

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

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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 the 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. The evaluation of release rate due to container lid gaps has been performed by CRIEPI and BAM. In CRIEPI, the gap of the flow path was calculated from the roughness of the container surface without a quantitative assessment of the severity of the accident. In this work, 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.

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

Development of Liquid Fuel Fabrication and Fission Product Management Technologies for Chloride-based Molten Salt Reactors

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Molten Salt Reactor (MSR) is one of Generation-IV nuclear reactors that uses molten salts as a fuel and coolant in liquid forms at high temperatures. The advantages of MSR, such as safety, economic feasibility, and scalability, are attributed from the fact that the molten salt fuel in a liquid state is chemically stable and has excellent thermo-physical properties. MSR combines the fuel and coolant by dissolving the actinides (U, Th, TRU, etc.) in the molten salt coolant, eliminating the possibility of a core meltdown accident due to loss of coolant (LOCA). Even if the molten salt fuel leaks, the radioactive fission products dissolved in the molten salt will solidify with the fuel salt at room temperature, preventing potential leakage to the outside.

MSR was first demonstrated at ORNL starting with the Aircraft Reactor Experiment (ARE) in 1954 and was extended to the 7.4 MWth MSRE developed in 1964 and operated for 5 years. Recently, various start-ups, including TerraPower, Terrestrial Energy, Moltex Energy, and Seaborg, have been conducting research and development on various types of MSR, particularly focusing on its inherent safety and simplicity. While in the past, fluoride-based molten salt fuels were used for thermal neutron reactors, recently, a chlorine-based molten salt fuel with a relatively high solubility for actinides and advantageous for the transmutation of spent nuclear fuel and online reprocessing has been developing for fast neutron spectrum MSRs.

This paper describes the development status of the process and equipment for producing high-purity UCl_3 , a fuel material for the chlorine-based molten salt fuel, and the development status of the gas fission product capturing technologies to remove the gaseous fission products generated during MSR operation. In addition, the results of the corrosion property evaluation of structural materials using a natural circulation molten salt loop will also be included.

Keywords: Molten salt reactor (MSR), U Chlorination, Fission product removal, Corrosion

Sensitivity Analysis of Model Parameters on the Fracture Resistance of SNF Cladding With Hydride

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The damage ratio of Spent Nuclear Fuel (SNF) is a very important intermediate variable for dry storage risk assessment which require an interdisciplinary and comprehensive investigation. 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 sensitivity analysis was performed to evaluate the importance of the damage parameters that need to be calibrated for the simulation of zircaloy-4 cladding failure using computational mechanics. The simulation model was generated from a microscopic image of the cladding with hydride. The image segmentation method was used to separate the Zircaloy-4, hydride, and hydride- Zircaloy matrix interfaces to create a pixel-based finite element model. The ring compression test (RCT) was simulated because the resistance of the cladding under pinch load can be evaluated by this test. It was assumed that the damage starts with the formation and growth of voids or small cracks in the material, which grow and combine to form larger cracks, eventually leading to the complete fracture of the material. Therefore, the ductile damage criterion was applied to all materials to simulate crack formation and propagation. The sensitivity analysis was performed based on the design of experiments using L8 orthogonal array. The effects of five factors on the fracture resistance of hydrided cladding were quantified, and they are the fracture strains describing the damage initiation in zircaloy-4 matrix, hydride, and hydride-zirconium matrix, and yield stress and Young's modulus for hydride-zirconium matrix. Information on those parameters are hardly available in literature and experimental data which enable the estimation of those are also very rare. It is planned to build a computational model which can accurately simulate the fracture behavior of hydrided cladding by calibrating significant fracture parameters using reverse engineering. The results of this study will help to figure out those significant parameters.

Keywords: Spent nuclear Fuel, Hydrided cladding, Damage parameter, Sensitive analysis, Fracture resistance, Ring compression test

Neutron Detection and Counting According to Burnup History Using MCNP

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Recently, the spent fuel pools withdrawn from nuclear power plants in Korea have been saturated. Therefore, specific regulations on the management of spent fuel pools, such as transportation and intermediate storage are needed. The burnup history is directly related to the management of spent nuclear fuel. This is because the decision to handle nuclear fuel may vary depending on the initial concentration of nuclear fuel, the degree to which nuclear fuel is irradiated and radioisotope nuclides are decayed, and the cooling state in the spent nuclear fuel storage tank.

The purpose of this study is to determine the burnup of fuel based on the value obtained by scanning the surface of spent nuclear fuel through a neutron detector. Conversely, a database of neutron signals that scan bundles of spent nuclear fuel with an instrument with an already identified combustion history needs to be completed. First of all, the correlation between burnup history and nuclides was identified in previous studies. By setting the burnup history as the input value in the ORIGEN-ARP code, it was possible to identify the radioactive isotopes remaining in the bundle of nuclear fuel. Neutrons can finally be measured based on the amount of nuclide inventory that constitutes spent nuclear fuel. Through MCNP, the neutron detector was simulated and signals were measured to confirm how it correlates with the previously acquired burnup history database. In addition, the M (sub-critical multiplication) value, which is essential for neutron measurement, was checked to confirm the degree to which additional neutrons were generated in spent nuclear fuel in a subcritical state. The target nuclear fuel assembly was CE16×16, WH14×14, and WH17×17, which confirmed the correlation (1) between burnup, enrichment, and cooling time with the previous research topic, TNSI (Total neutron source intensity).

$$BU(GWd/MTU) = 0.83(S_N)^{0.238} \cdot (IE)^{0.4627} \cdot e^{0.0083 \cdot t} \quad (1)$$

A neutron signal will be obtained from the case according to each burnup history constituting this database. In particular, PAR=SF, a function that calculates the production amount of the fission product, was used. To confirm the computational logic of SF, it was confirmed whether a reasonable calculation was made by calculating with a nuclide spectrum.

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Keywords: Spent fuel, MCNP, Detection, Neutrons, Burnup history

2분과

사용후핵연료 처분전관리(Poster)

Verification of Burnup of Spent Nuclear Fuel Discharged From OPR1000 NPPs

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One of the most important factors in the delivery and acceptance requirements for dry storage of spent fuel is the burnup of spent fuel. Here, burnup has a unit of MWD/MTU and is used as a measure of how much nuclear fuel is depleted in a nuclear reactor. In addition, since it is one of the most basic characteristic information for the soundness evaluation of spent nuclear fuel, it is a required item not only by regulatory agencies but also by KORAD, the acquiring agency. The burnup of spent nuclear fuel is the burnup calculated through flux mapping using signals measured from in-reactor instruments during nuclear power plant operation (hereinafter: actual burnup) and the burnup calculated using the core design code (hereinafter: design burnup). In this paper, the design burnup of spent nuclear fuel discharged from OPR100 NPPs (Nuclear Power Plants) in Korea was recalculated to confirm the reliability of the actual burnup currently managed at the nuclear power plant. Basically, since spent nuclear fuel must maintain subcriticality under wet storage or dry storage, a burnup error of about 5% is considered as a conservative approach when evaluating the criticality safety of wet storage tanks and dry storage systems. Therefore, in this paper, we tried to verify whether the difference between actual burnup and design burnup for all spent nuclear fuel released from domestic OPR100 type light water reactor nuclear power plants is within 5%. As a result of the evaluation, the largest deviation between actual burnup and design burnup was about 1,457 MWD/MTU, and when converted into a percentage, it was about 3.3%. Therefore, it was confirmed that the actual burnup managed by OPR1000 NPPs in Korea has sufficient reliability. In the future, we plan to check the reliability of the performance burnup managed in WH NPPs, and some of them will be verified through measurement.

Keywords: Spent fuel, Dry storage, Safety information, Burnup, OPR1000

Two Step Process for the Recovery of Strontium From Salt Solution

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It has been investigated on the management of the nuclides in KAERI. Strontium-90 is a high heat-generating nuclide in spent nuclear fuel. It is needed to separate the salt 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. It was investigated on operating conditions of reactive distillation process for the recovery of the strontium from the salt solution. At a reduced pressure, considerable amount of the carbonation agents such as K_2CO_3 and Li_2CO_3 were reduced during heating in the distiller due to the thermal decomposition. Therefore, the two step process was proposed, which is composed of a reaction step at an atmospheric pressure and a salt distillation step at a reduced pressure. In the reaction step, the condition of low temperature and high pressure is suitable to suppress the decomposition of the carbonation agent. In the salt distillation step, reduced pressure is preferable at a suitable temperature depending on the evaporation rate of the salt.

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Keywords: Strontium, Chlorination, Recovery, Molten salt

Analysis of the Mixed Gas Generated on Carbon Anode During the TiO Reduction Reaction

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Instead of using expensive platinum, carbon anodes could potentially be utilized in the process of reducing oxides in LiCl-Li₂O molten salt at a high cell potential. However, this high potential leads to the generation of a mixture of anodic gases containing toxic and corrosive gases such as chlorine (Cl₂), oxygen (O₂), carbon monoxide (CO), and carbon dioxide (CO₂). To better understand this gas mixture, we conducted real-time analyses of the gases generated on the carbon anode during the TiO reduction reaction in the molten salt at 650°C, using a MAX-300-LG gas analyzer. Our results indicate that the ratio of CO/O₂/CO₂/Cl₂ in the gas mixture is significantly influenced by the composition of the salt, and that removing the sources of oxygen ions in the salt increases the likelihood of generating toxic and corrosive Cl₂ gas.

Keywords: Anodic gas analysis, Oxide reduction reaction, Carbon anode, Pyroprocessing, Titanium monoxide

Nuclide Separation From the Simulated Oxide Fuel With MgCl_2 at 800°C

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Separating nuclides from spent nuclear fuel is crucial to reduce the final disposal area. The use of molten salt offers a potential method for nuclide separation without requiring electricity, similar to the oxide reduction process in pyroprocessing. In this study, a molten salt leaching technique was evaluated for its ability to separate nuclides from simulated oxide fuel in MgCl_2 molten salts at 800°C . The simulated oxide fuel contained 2wt% Sr, 3wt% Ba, 2wt% Ce, 3wt% Nd, 3wt% Zr, 2wt% Mo, and 89wt% U. The separation of Sr from the simulated oxide fuel was achieved by loading it into a porous alumina basket and immersing it in the molten salt. The concentration of Sr in the salt was measured using ICP analysis after sampling the salt outside the basket with a dip-stick technique. The separated nuclides were analyzed with ICP-OES up to a duration of 156 hours. The results indicate that Ba and Sr can be successfully separated from the simulated fuel in MgCl_2 , while Ce, Nd, and U were not effectively separated.

Keywords: MgCl_2 molten salt, Nuclide separation, Simulated oxide fuel, Molten salt leaching technique

Compression Test of Simulated Fuel

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When damaged nuclear fuel is stripped and re-fabricated into stabilized pellets, it is necessary to analyze the characteristics of the stabilized pellets, such as density, leaching behavior, and compressive strength, for final disposal.

In this study, simulated nuclear fuel with UO_2 and burn-up of 35 GWd/tU and 55 GWd/tU was used to measure the compressive strength of the stabilization pellet.

In order to change the density of the sintered pellet, a sintered pellet was prepared by heat treatment at 1,550°C and 1,700°C for 6 hours in a reducing atmosphere of 4% H_2/Ar . In the case of UO_2 , the density was 10.4 g/cm³ (94.5% of T.D.) and 10.6 g/cm³ (96.6% of T.D.) depending on the sintering temperature (1,550°C, 1,700°C). In the case of simulated fuel with a burn-up of 35 GWd/tU, the density was 8.8 g/cm³ (80.9% of T.D.) and 10.2 g/cm³ (93.6% of T.D.) depending on the sintering temperature (1,550°C, 1,700°C). In the case of simulated fuel with a burn-up of 55 GWd/tU, the density was 8.3 g/cm³ (77.0% of T.D.) and 10.0 g/cm³ (92.3% of T.D.) depending on the sintering temperature (1,550°C, 1,700°C). It was found that the compressive strength of simulated nuclear fuel decreased with increasing burn-up and increased with increasing density. In the case of UO_2 , the compressive strengths were 717.8 MPa and 897.4 MPa when the densities were 10.4 g/cm³ and 10.6 g/cm³, respectively. In the case of simulated nuclear fuel with a burn-up of 35 GWd/tU, the compressive strengths were 472.1 MPa and 732.3 MPa when the densities were 8.8 g/cm³ and 10.2 g/cm³. In the case of simulated nuclear fuel with a burn-up of 55 GWd/tU, the compressive strengths were 301.4 MPa and 515.5 MPa when the densities were 8.3 g/cm³ and 10.0 g/cm³, respectively.

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Keywords: Compressive strength, Simulated fuel, Burn-up, Density, Sintered pellet

Design Requirements of Dry Storage Inspection System for CANDU Spent Fuel Storage Integrity

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Spent fuel from the Wolsong CANDU reactor has been stored in above-ground dry storage canisters. Wolsong concrete dry storage canisters (silos) are around 6 m high, 3 m in outside diameter, and have shielding comprised of around 1 m of concrete and 10 mm 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 304 L stainless steel, around 42" in diameter, 22" in height, and hold 60 fuel bundles each. The operating license for the dry storage canisters needs to be extended. It is desired to perform in-situ inspections of 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. KHNP-CNL (Canadian Nuclear Lab.) has set-up the design requirements to perform the in-situ inspections in the dry storage canisters. This Design Requirements applies to the design of the dry storage canister inspection system.

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Keywords: CANDU spent fuel, Dry storage inspection system, Design requirements, Silo, Canister

Thermal Analysis of Single Spent Fuel Assembly Under Different Gas Backfills

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Spent nuclear fuels released from the reactor are stored in cooling pools and then stored in dry storage casks. During the transition from the wet storage to dry storage cask, a vacuum drying process is used to remove residual water in the cask. During the vacuum drying process, gas pressure is reduced to below 400 Pa to promote evaporation and water removal. KAERI is developing a PWR single assembly (PLUS7) test equipment to simulate the thermal flow in spent fuel assembly.

In this study, the thermal conductivity of air at low pressure was derived to perform the thermal analysis of the canister in vacuum. In addition, thermal analyses were performed for the canister with backfill gases of helium, air, and a vacuum in the vertical orientation using the COBRA-SFS code.

At low pressure, the thermal conductivity of air depends on pressure and temperature. The reduced thermal conductivity, k_r (W/m-K) was calculated using the curve fit for air at reduced pressure in thin gaps presented in the General Electric Fluid Flow Handbook. $k_r/k_0 = \frac{1}{1 + CT/P\delta}$

Where, k_0 is the thermal conductivity at atmospheric pressure (W/m-K), P is the reduced (vacuum) pressure (Pa), δ is the gap size (m), T is the temperature (K), and C is the Lasance constant ($7.657E-5$ N/m-K). The thermal conductivity of air decreases as the pressure decreases. The reduced thermal conductivity of air at pressures of 400 Pa and 40 Pa was calculated to be 0.97 and 0.77, respectively. For the analysis in vacuum, no enhancement of the convective heat transfer was assumed ($Nu=1.0$).

For the helium backfill, the peak cladding temperature was the lowest and the axial temperature profile was the flattest due to the higher thermal conductivity and lower density of the helium. For the vacuum backfill, the peak cladding temperature was the highest and temperature gradient was the sharpest due to the only radiative heat transfer effect in the fuel assembly.

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Keywords: Thermal analysis, Spent fuel, Storage cask, Vacuum drying, Thermal conductivity, Convective heat transfer, Peak cladding temperature, Radiative heat transfer

Fuel Material Recovery From Simfuel Rod-Cuts by Oxidative Decladding

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The damaged spent fuel rods must be stabilized by encapsulation or dry re-fabrication technologies before geological disposal. For applying the dry re-fabrication technology, we manufactured a vertical type furnace to perform the fuel material recovery from damaged fuel rods by oxidative decladding technology. As driving forces to accelerate oxidative decladding rate, magnetic vibration and pulse hammering generated by a pneumatic cylinder were used in this study. The oxidative decladding efficiency and recovery rate of fuel oxide powder with rod-cut length, oxidation temperature and time, oxygen concentration, and gas mixtures were investigated using simfuel rod-cuts in a vertical furnace for fuel material recovery and powder quality improvement. The oxidative decladding was performed for 2.5-10 h as following operation parameters: simfuel rod-cut length of 50-200 mm, oxidative temperature from 450 to 580°C, oxygen concentration of 49.5 or 75.6%, and gas mixtures in O₂/Ar or O₂/N₂. In magnetic vibration, oxidative decladding was progressed only at bottom portion of fuel rod-cut. Whereas, oxidative decladding in pulse hammering was occurred at both top and bottom portions of fuel-rod. In pulse hammering method, the oxidative decladding conditions to declad rod-cuts of 50-200 mm in length were established to achieve both decladding efficiency of ~100% and fuel material recovery rate of > 99%. These conditions were as follows: oxidation temperature and time at 500°C and 2.5-10 h, oxygen concentration at 75.6% under O₂/N₂ gas mixtures. As operation conditions for a pneumatic cylinder, stroking, actuating, and waiting times were 0.5, 3, and 12 s.

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Keywords: Fuel material, Oxidative decladding, Vertical furnace, Vibration, Hammering

Preliminary Preparation of Bead Sorbent for Iodine Capture

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The Ag^0 -containing sorbents synthesized by Na, Al, and Si alkoxides have higher maximum iodine capture capacity and textural properties than zeolite-based Ag^0 -containing sorbents. However, these sorbents were prepared in the form of granules via a step for cutting cylindrical alcogels. Since as-made sorbents decreased packing density, they must be additionally crushed and then classified into an appropriate size for increasing packing density. The bead formation in the step of sol-gelation could bring about the simplification of sorbent preparation process and an improvement of packing density. In the Na, Al, and Si alkoxides as starting materials, sol solution was hydrophilic and lower density than vegetable oil, which transformed sol droplets to sol-gel beads. Thus, in these precursors, sol droplets, which must be sprayed by single nozzle placed at bottom side of oil column, can rise up through oil column. Acetic acid (HOAc) was used as the catalyst for the hydrolysis of Na alkoxide (TEOS) and gelation of the $\text{Na}^+\text{AlSi-OH}$ alcogel. For obtaining sol-gel beads, experiments were performed by the flowrate change of sol solution and HOAc at different nozzle sizes using soybean oil column of 1 m in length. At a sol/HOAc flowrate ratio of 3.85, some $\text{Na}^+\text{AlSi-OH}$ alcogel beads were obtained. After the Ag/Na ion-exchange, Ag content in $\text{Ag}^+\text{AlSi-OH}$ hydrogel was low due to reaction between Na^+ and HOAc during sol-gelation and aging step. The $\text{Ag}^+\text{AlSi-OH}$ hydrogel with high Ag content could be prepared by Na addition. After the solvent exchange and drying at ambient pressure, the bead sorbents had higher Ag^0 content and larger pore size than granular sorbents. However, further experiments are needed to increase yield rate in bead sorbent.

Acknowledgements

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Keywords: Iodine capture, Bead formation equipment, Sol droplet rising, Sol bead

Leaching Behavior of Stabilized SIMFUEL and UO₂ Pellets in Water

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The stabilization techniques are highly required for damaged nuclear fuel to strengthen safety in terms of transportation, storage, and disposal. This technique includes recovering fuel materials from spent fuel, fabrication of stabilized pellets, and fabrication of fuel rods. Thus, it is important to identify the leaching behavior of the stabilized pellets to verify their stability in humid environments which are similar to storage conditions. In this study, we introduce various leaching experiment methods to evaluate the leaching behavior of the stabilized pellets, and determine the most suitable leaching test methods for the pellets. Also, we establish the leaching test conditions with various factors that can affect the dissolution and leaching behavior of the stabilized pellets. Accordingly, we prepare the simulated high- (55 GWd/tU) and low- (35 GWd/tU) burnup nuclear fuel (SIMFUEL) and pure UO₂ pellets sintered at 1,550°C and 1,700°C, respectively. Each pellet is placed in a vessel and filled with DI water and perform the leaching test at three different temperature to verify the leaching mechanism at different temperature range. Based on the standard leaching test method (ASTM C1308-21), the test solution is removed from the pellet after specific time intervals and replaced in the fresh water, and the vessel is placed back into the controlled-temperature ovens. The test solutions are analyzed by using ICP-MS.

Keywords: Damaged spent fuel, Leaching, SIMFUEL, UO₂, PCT-A, ASTM C1308

Long-Term Structural Integrity Evaluation for Wolsong Silo Storage Systems

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On-site storage facility using concrete silo dry storage systems for spent nuclear fuel at Wolsong NPP site came into operation in 1992 and was expanded four times, and a total of 300 silo dry storage systems are currently in operation. The design lifetime of silo dry storage systems has been licensed for 50 years. As the dry storage systems are subject to time constraints for a limited lifetime, countries operating the dry storage systems are working to ensure the long-term integrity of dry storage systems and IAEA also recommends that the dry storage systems be assessed for long-term storage.

To demonstrate the long-term integrity due to material degradation during the licensed design lifetime, the structural integrity of silo dry storage systems was evaluated by considering the material degradation characteristics of concrete. The concrete compressive strength results measured so far by the rebound hammer method, which is an internationally standardized nondestructive test method for converting hardness into compressive strength using the correlation between rebound number and strength at the time of a Schmidt hammer strike, were analyzed in accordance with Wolsong NPP's procedure to quantify the degradation characteristics, and the prediction of concrete strengths for 20 years and 50 years after construction of the silo dry storage systems was determined, respectively.

Based on these residual compressive strengths, structural analyses of the silo dry storage systems were carried out under normal, off-normal and accident conditions of the related regulations, and the structural integrity of silo dry storage systems was reevaluated. It was confirmed the silo dry storage systems are able to maintain structural integrity up to the design lifetime of 50 years even if the concrete is deteriorated.

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Keywords: Silo dry storage system, Design lifetime, Concrete compressive strength, Long-term integrity

Burnishing Treatment to Prevent Chloride-Induced Stress Corrosion Cracking of Welded Canisters

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The spent fuel storage canister is generally made of austenitic stainless-steel and has the role of an important barrier to encapsulate spent fuels and radioactive materials. Canister near coastal area has welding lines, which have high residual tensile stresses after welding process. Interaction between austenitic stainless steel and chloride environment forms detrimental condition causing chloride induced stress corrosion cracking (CISCC) in canister. Reducing or eliminating tensile stress on canister can significantly decrease probability of crack initiation. Surface stress improvement works by inducing plastic strain which results in elastic relaxation that generates compressive stresses. Surface stress improvement methods such as burnishing process can effectively prevent for CISCC of canister surfaces. In this study, burnishing treatment has been evaluated to control residual tensile stress practically applicable to atmospheric CISCC for aging management of steel canisters.

Burnishing process was selected as a prevention technology to CISCC of stainless steel canisters to improve resistance of CISCC through enhancement of surface roughness and generation of compressive residual stress. SUS 316 SAW (Submerged Arc Welding) specimens were burnished with flat roller and round roller after manufactured and assembled on CNC machine using base plate.

The burnishing test results showed that the surface roughness of SUS 316 SAW welded specimens after roller burnishing of pass No. 5 was improved with 85% with flat roller and 93% with round roller, individually. Surface roughness showed the best state when burnished at pressure of 115 kgf, feeding rate of 40 $\mu\text{m}/\text{stroke}$ and pass No. of 5 turns with round roller. The surface of SUS 316 SAW welded specimens had much high residual compressive stress than yield stress of SUS 316 materials with roller burnishing treatment, independently of kinds of roller. The surface of the welded specimen by round roller burnishing showed smaller compressive stress and deeper stress region than in the surface of flat round roller burnishing.

The roller burnished canister with good surface roughness could reduce the number of crack initiation sites and the high residual compressive stress formed on the welded surface might prevent the crack initiation by reducing or eliminating tensile residual stress in the weld zone, finally leads to excellent CISCC resistance. The crack growth behavior of SUS 316 welded specimens will need to investigate to evaluate the corrosion integrity of the canister materials under chloride atmosphere according to burnishing treatment.

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

Upgrade and Modification of Fuel Handling Equipment at Operating Nuclear Power Plants

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The purpose of this study is to provide technical issues in upgrade and modification of fuel handling equipment at operating nuclear power plants. The improvement for safety function and performance enhancement of fuel handling equipment has been going on for 20 years since the early 2000's. This improvement is recently focused on the replacement of components through the performance analysis and the operation and maintenance plan based on replacement cycle of its component. Additionally, it is required to secure spare parts so that it can be operated at all times with compatibility and standardization to other domestic nuclear power plants.

The fuel handling equipment is consisted of refueling machine, upender and carriage of fuel transfer system, spent fuel handling machine, new fuel elevator and various tools, and the equipment are linked in systematic interlocks. Fuel handling is a critical task during a nuclear power plant refueling outage. Even minor component defects may stop operation of the whole system and have a significant impact on the overall system process. To achieve this goal, major components that are expected to be replaced for reliable operation are summarized as follows; 1) motor assembly with AC servomotors and driver for bridge, trolley and hoist of refueling machine and spent fuel handling machine, 2) winch motor and drive for upender and carriage of fuel transfer system, 3) operator control console with a HMI PC base PLC (Programmable Logic Controller) control system, 4) positioning and load weighing sensors such as an encoder and a load cell with its support for periodic calibration and maintenance, 5) main power draped style festoon cable assembly for bridge of refueling machine, 6) pneumatic control assembly for gripper operation of refueling machine, 7) active components (e. g., air motor, hydraulic cylinder and limit switch) to be removable and re-installable without requiring the water level to be lowered.

It is advisable to utilize such various information as it can help to improve reliability of fuel handling as a critical path in upgrade and modification of fuel handling equipment at operating nuclear power plants.

Keywords: Fuel handling equipment, Upgrade and modification, Refueling equipment

Review of Spent Fuel Pool Accident Cases in Foreign Countries

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After Fukushima nuclear power plant accident in 2011, Concerns about accident of spent fuel pool increase. In Korea, the time of saturation of spent fuel pool is coming, but regulatory measures and safety evaluation are insufficient when occurring spent fuel pool accident. Thus, it is necessary to review of spent fuel pool accident in foreign countries to establish regulatory measures and safety evaluation of spent fuel pool accident suitable for domestic spent fuel pool. Therefore, we reviewed spent fuel pool accident that occurred at Fukushima Unit 4, SONGS Unit 2 and PAKS. In Japan, spent fuel pool accident occurred at Fukushima NPP in 2011. Tsunami was cause of the accident. Station Black Out occurred at Fukushima NPP and Emergency Diesel Generator lost their functions due to Tsunami. As a result, Loss of cooling happened in spent fuel pool at Fukushima NPP. For Unit 4, wall of spent fuel pool in Unit 4 was damaged due to hydrogen explosive, so loss of coolant in spent fuel pool of Unit 4 occurred. After the accident, the temperature of spent fuel pool increases to 75°C, but there was no damage to the spent fuel. In USA, spent fuel pool accident occurred at SONGS Unit 2 in 2013. The debris of nearby ocean is cause of the accident. The debris entered the system through a damaged Salt Water Cooling pump suction strainer. The debris obstructed flow through the Component Cooling Water heat exchanger and operation of Salt Water Cooling. The maximum spent fuel pool temperature during this event was 25.6°C. It was a value that satisfied the technical specifications of the SONGS NPP. In Ukraine, spent fuel pool accident occurred at PAKS in 2003. Unintentionally opened valve of cleaning tank is cause of the accident. Loss of coolant occurred in spent fuel pool of PAKS. Due to loss of coolant, spent fuels were exposed to the vapor state atmosphere, and oxidation occurred in the cladding tube of the spent fuel that rose to 1,400°C. In this study, Review of spent fuel pool accident in major foreign countries was conducted as basic studies for establishing regulatory measures and safety evaluation of spent fuel pool in Korea. Causes of each accident were different by structure of spent fuel pools. Result of this study will be contributed to establish safety measures of spent fuel pool accident suitable for domestic spent fuel pool facility.

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Keywords: Spent nuclear fuel, Spent fuel pool, Storage facility, Wet storage

Selective Chlorination of Actinide and Alkali Earth Elements in Spent LWR Nuclear Fuel Using Ammonium Chloride

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As temporary storage facilities for spent nuclear fuel (SNF) are becoming saturated, there is a growing interest in finding solutions for treating SNF, which is recognized as an urgent task. Although direct disposal is a common method for handling SNF, it results in the entire fuel assembly being classified as high-level waste, which increases the burden of disposal. Therefore, it is necessary to develop SNF treatment technologies that can minimize the disposal burden while improving long-term storage safety, and this requires continuous efforts from a national policy perspective. In this context, this study focused on reducing the volume of high-level waste from light water reactor fuel by separating uranium, which represents the majority of SNF. We confirmed the chlorination characteristics of uranium (U), rare earth (RE), and strontium (Sr) oxides with ammonium chloride (NH_4Cl) in previous study. Therefore, we prepared U-RE-SrO_x simulated fuel by pelletizing each elements which was sintered at high temperature. The sintered fuel was again powdered by heating under air environment. The powdered fuel was reacted with NH_4Cl to selectively chlorinate the RE and Sr elements for the separation. We will share and discuss the detailed results of our study.

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

Dissolution of ZrCl_4 in the LiCl-KCl Molten Salt

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As a method for chlorinating spent nuclear fuel, a method of using ZrCl_4 in high-temperature molten salt is known. However, ZrCl_4 has a sublimation property that vaporizes at a temperature similar to the melting temperature of molten salt. Since solubility of ZrCl_4 in molten salt is very low, it is difficult to dissolve a large amount of ZrCl_4 in molten salt. However, once ZrCl_4 can be dissolved together with the molten salt, it remains in the molten salt without vaporizing. That is, it is known that when vaporized ZrCl_4 reacts with molten salt in a sealed reactor, it dissolves into the molten salt, and ZrCl_4 above the solubility remains in the molten salt in the form of M_2ZrCl_6 . Here, M represents an alkali element.

Therefore, in this study, a flange-type sealed reactor was fabricated to dissolve a large amount of ZrCl_4 in LiCl-KCl salt, and LiCl-KCl salt in which ZrCl_4 was dissolved as K_2ZrCl_6 was prepared. LiCl-KCl , KCl , and ZrCl_4 salts were charged into alumina crucibles and placed in a sealed reactor. The reactor was heated to 500°C and the reaction time was about 20 hours. The temperature of the reactor surface was about 480°C . After completion of the reactions, each crucible was recovered from the inside of the reactor. All of the ZrCl_4 vaporized and there was no residue in the crucible. Both KCl and LiCl-KCl salts appear to have dissolved and then cooled, with respective weight gains. XRD analysis was performed to observe the structure of the recovered salts, and ICP analysis was performed to measure the Zr element content in each salt. As a result of XRD analysis, the structure of K_2ZrCl_6 was found in the KCl salt, but not in the LiCl-KCl salt. As a results of ICP analysis, it was found that the LiCl-KCl salt contained about 33wt% of ZrCl_4 , and about 25wt% was dissolved in the KCl salt. In other words, it was shown that ZrCl_4 above the solubility can be dissolved in the LiCl-KCl molten salt.

Keywords: Zirconium chloride, Molten salt, Dissolution, Chlorination

A Study on the Long-term Cooling Effect of Hydride Reorientation in Unirradiated Zircaloy-4 Cladding (3, 6 & 12 Months)

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The hydride reorientation (HR) of the post-irradiated nuclear fuel cladding after use affects the integrity of the spent nuclear fuel. During the dry storage process, which is an intermediate storage method, it was found that the hydride in the circumferential direction is rearranged into radial hydride, and this is believed to be due to factors such as hoop stress, peak temperature, accumulated hydrogen concentration, and cooling rate during the storage period.

$$f(\text{HR}) = f(T_{\text{max}}) + f(\sigma_H) + f(C_H) + f(\Delta T) + f(10\text{Cy}) + f(\text{cooling rate}) + \dots$$

To simulate long-term dry storage of spent nuclear fuel, the hydride reorientation behavior was evaluated using unirradiated Zircaloy-4 (CWSRA) cladding with hydrogen charged under various hoop stresses (70, 80, 90, and 110 MPa) at long-term cooling periods (3, 6, and 12 months). Test results showed that as the cooling time increased, the sample with 90 MPa hoop stress at a maximum temperature of 400°C approached the ductility recommendation limit of 2%.

In a 90 MPa hoop stress specimen with 3 months cooling period at peak temperature of 400°C, the offset strain was 4.24% at room temperature RCT, while it showed the result of 2.86% for the cooling period of 12 months. On the other hand, the specimen with hoop stress of 110 MPa and cooling period of 12 months showed result of 1.4%.

The test results need to take into account errors in hydrogen charging and hydrogen analysis, and it is necessary to consider reproducibility through repeated tests. These results indicate the need for continued attention to the evaluation of the effects of hydride reorientation due to long-term cooling in the context of the integrity of spent fuel.

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Keywords: Hydride reorientation, Long-term cooling, Hydrogen concentration, Radial hydride

A Study on the Behavior of Low-concentration Hydrogen on the Hydride Reorientation of Unirradiated Zircaloy-4 Cladding

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In the process of spent fuel dry storage, which is an intermediate management method, it was found that hydrides in the circumferential direction rearranged into radial hydrides. Various factors, such as hoop stress, peak temperature, cooling rate during the storage period, and hydrogen concentration accumulated during the burnup process, significantly affect the susceptibility of spent fuel cladding.

In recent studies based on the hydrogen solubility value of about 210 ppm corresponding to the peak temperature of 400°C, if the threshold stress decreases as the hydrogen concentration increases in the low hydrogen range under 210 ppm, the threshold stress increases as the hydrogen concentration increases in the low hydrogen range under 210 ppm. The fundamental cause of this trend is the diffusion of hydrogen into the high-stress region due to the stress gradient formed in the specimen, and hydrogen compounds which remain undissolved in the circumferential direction, even at the peak temperature, play a crucial role to determine the magnitude of the threshold stress.

This study evaluated the behavior of hydride reorientation under various hoop stress conditions (70, 80, 90, and 110 MPa) using unirradiated Zircaloy-4(CWSRA) cladding tubes under long-term cooling conditions (3, 6, and 12 months). The results of analyzing the offset strain by hydrogen concentration for long-term cooling showed that specimens with low hydrogen concentration exhibited higher integrity than specimens with high hydrogen concentration at hoop stresses of 90 and 110 MPa.

The HR test using irradiated fuel cladding showed that specimens with low hydrogen concentrations exhibited relatively higher susceptibility. To quantify these results, it is necessary to research further in detail by repeated tests.

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Keywords: Hydride reorientation, Long-term cooling, Threshold hoop Stress, Radial hydride

Explicit CFD Analysis Model Validation 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. During long-term storage, there are several mechanisms that could result in the degradation of spent nuclear fuels, and the temperature is the most important parameter to predict and estimate the degradation behaviors. Therefore, 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.

Herein, Explicit CFD analysis model is introduced and validated by estimating the thermal hydraulic response of the dry storage system that is Dry Cask Simulator (DCS). Extended Storage Collaborating Program (ESCP) led by the Electric Power Research Institute (EPRI) is organized to assess 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 is used to simulate DCS, and the test condition of 0.5 kW decay heat and 100 kPa helium pressure was investigated in this study. In case of peak cladding temperature (PCT), PCT from the experiment was 376 K while that of CFD was 374 K. It implies CFD simulation gives good agreement with experimental measurement. Peak temperatures of channel can, basket, canister and shell predicted by CFD simulation also show good prediction and the discrepancies were less than 7 K while measurements uncertainty was 7 K.

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

Preliminary Study on the Framework to Estimate Safety Significance of Components in Dry Storage Systems for Spent Nuclear Fuel

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Once systems, structures and components (SSCs) of dry storage systems are classified with respect to safety function or safety significance (i.e., safety classification), appropriate engineering rules can be applied to ensure that they are designed, manufactured, maintained, managed (e.g. aging management) etc.

In United States, the systems, structures and components (SSCs) consisting DSSs are classified into two or several grades (i.e., class A, B and C or not important to safety, and important to safety (ITS) or not important to safety (NITS)) with respect to intended safety function and safety significance. This classification methods were based on Regulatory Guide 7.10 (i.e., guidance for use in developing quality assurance programs for packaging). Also, in Korea, SSCs of DSSs should be classified into ITS and NITS in much the same as method based on Regulatory Guide 7.10. In that guidance, for providing graded approach to manage the SSCs of packaging, they were trying to classifying SSCs in accordance with radiological consequences. But there was limitations that the provided classification criteria was still qualitative, so that it was not enough for managing the SSCs according to graded approach.

On the other hand, in some other nuclear facilities (i.e., nuclear power plant, radioactive waste management facility and disposal facility etc.), quantitative criteria relevant to radiological consequence (i.e., radiation doses to workers or to the public) or inventory of radioactivity are existed so that it can be applied for classifying safety classes. In summary, the study on the application safety classification that applied quantitative criteria to perform safety classification of SSCs in DSS is inadequate or insufficient.

The purpose of this study is proposing the preliminary framework for estimating safety significance of SSCs in DSS which can be utilized in our further advanced studies. In this study, a framework was established to estimate the safety significance of SSCs related to radiation shielding and confinement using MCNP[®] 6.2 and Microsoft Excel. Referring to the methodology of IAEA Specific Safety Guide 30, we assumed severity for failures of components that could lead to degradation of the SSC's performance. The safety class of SSC was decided based on the impact of SSC's failure on consequences.

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

Analysis of Basic Characteristics for Spent Fuel

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The spent fuel is classified based on the arrangement of fuel rods, which is considered the primary characteristic data for selecting nuclear fuel. The reason for prioritizing the classification by fuel rod arrangement is that it has the greatest physical impact on the production, supply, operation, reactor type, rack size within the containment vessel, and specifications for the basket in the future dry storage system. Additionally, as mentioned earlier, various meanings of nuclear fuel types are distinguished according to the arrangement of fuel rod. The burnup and cooling period ranges are also important factors in the characterization analysis for the selection of spent fuel, the burnup range was set for both low and high burnup ranges and the cooling period is necessary to consider the reliability during handling of nuclear fuel thermal distribution within the storage system

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Keywords: Spent fuel characteristic, Classification, Fuel rod arrangement, Burnup range, Cooling period

Analysis of Receipt and Storage Procedures for Spent Nuclear Fuel in Dry Storage Facilities Based on Concrete Overpack

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In Korea, the construction of dry storage facilities for spent nuclear fuel is being promoted through the 2nd basic plan for high-level radioactive waste management. When operating dry storage facilities, exposure dose assessment for workers should be performed, and for this, exposure scenarios based on work procedures should be derived prior. However, the dry storage method has not yet been sufficiently established in Korea, so the work procedure has not been established. Therefore, research is needed to apply it domestically based on the analysis of spent nuclear fuel management methods in major overseas leading countries. In this study, the procedure for receiving and storing spent nuclear fuel in a concrete overpack-based storage facility was analyzed. Among the various spent nuclear fuel management systems, the metal overpack-based HI-STAR 100 system and the concrete overpack-based HI-STORM 100 system are quite common methods in the United States. Therefore, in this study, work procedures were analyzed based on each final safety analysis report. First, the HI-STAR 100 overpack enters the facility and is placed in the transfer area. Remove the impact limiter of the overpack and install the alignment device on the top of the overpack. Place the HI-TRAC, an on-site transfer device, on top of the alignment unit and remove the lids of the two devices to insert the canister into the HI-TRAC. When the canister transfer is complete, reseal the lid to seal it, and disconnect the HI-TRAC from the HI-STAR 100. Raise the canister-loaded HI-TRAC over the alignment device on the top of the HI-STORM 100 overpack and remove the lids of the two devices that are in contact. Insert the canister into the HI-STORM 100 and reseal the lid. The HI-STORM 100 loaded with spent nuclear fuel is transferred to the designated storage area. In this study, the procedure for receiving and storing spent nuclear fuel in a concrete overpack-based storage facility was analyzed. The main procedure was the transfer of canisters between overpacks, and it was confirmed that HI-TRAC was used in the work procedure. The results of this study can be used as basic data for evaluating the exposure dose of operating workers for the construction of dry storage facilities in Korea.

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Keywords: Spent nuclear fuel, Dry storage, Receipt and storage procedures, Concrete overpack

Calculation of the Reference Value at the End of the Drying Operation According to the Condition of the Canister

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As if the wet storage of Spent Nuclear Fuel (SNF) becomes saturated, a transition from wet storage to dry storage could be required. The first process for dry storage is to move SNF from the wet storage into a canister for dry storage, and secondly perform a drying process to remove the moisture in the canister to prevent a potential impact such as deterioration of cladding or corrosion of the interior material.

Nuclear Regulatory Commission (NRC) accepts the conditions describing the adequate dryness state that remain below the pressure of 3 Torr for 30 minutes in the drying process. That is, the most pressure of water vapor that may exist inside the canister is 3 Torr. If it is maintained below 3 Torr, it can be determined that the dryness criterion is satisfied. Based on this, relative humidity and dew point trends can be identified. Relative Humidity (RH) is calculated by dividing the vapor pressure by the saturated vapor pressure. Here, if the vapor pressure is fixed at 3 Torr, which is the dryness criterion value, the relative humidity has a value according to the saturated vapor pressure. Saturated vapor pressure is a value that varies with temperature, so relative humidity varies with temperature. On the other hand, the dew point temperature has a value according to the water vapor pressure. Therefore, when the internal temperature of the canister is 120°C and the water vapor pressure is 3 Torr, the relative humidity is 0.2% and the dew point temperature is -4.4°C.

We will confirm the suitability of the dryness criterion through the drying tests, and secure a technology that can measure and evaluate the amount of moisture remaining inside the canister.

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Keywords: Residual moisture, Dry storage, Spent nuclear fuel, Drying process

Development of Dynamic Leak Rate Evaluation Methodology for Loss of Confinement of Spent Fuel Dry Storage Systems

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Since the time to consider when evaluating leakage of spent fuel dry storage systems is very long, assumptions that continue to leak at the initial leakage rate are too conservative. Therefore, this study developed a dynamic methodology to calculate the change in leakage rate using time-varying variables and apply it to calculate the amount of radioactive leakage during the evaluation period. The developed dynamic methodology was then applied to calculate the leakage radiation source term for a hypothetical dry storage system and used to perform a public dose assessment.

When applying the developed dynamic leakage rate evaluation methodology for more accurate confinement evaluation in case of containment damage of dry storage system, it was found that the change of leak rate with time is very insignificant if the hole diameter is small enough, and the leak rate decreases rapidly with time when a hole with a certain diameter or larger occurs. In the case of the accident condition, except when the hole is very large, it corresponds to the choked flow condition, and the leak rate decreases rapidly as soon as the internal pressure is sufficiently lowered to enter the molecular and continuum flow region. In the case of a small hole diameter, the leakage volume is very small, so even if the dynamic methodology is applied, the evaluation results are not different from the case where the initial leakage rate continues, and when the hole diameter exceeds a certain value, the internal pressure drops according to the leakage volume, and the leakage rate decreases significantly. As a result of evaluating the dose to residents by applying the calculated radiation source term, it was confirmed that the dose criteria was exceeded when a hole with a diameter of about 4 μm occurred under off-normal conditions, and the dose standard was exceeded under accident conditions when a choked flow occurred between the diameter of the hole and 2-3 μm , resulting in a rapid increase in the dose. The results of this study are expected to contribute to a more accurate evaluation of the confinement performance of storage systems, which will contribute to the design of optimal dry storage systems.

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Keywords: Spent fuel, Dry storage system, Confinement evaluation, Public dose evaluation, Leak rate

Management and Disposal Plans for Spent Fuel/HLW in Various Countries Around the World

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As of 2023, there has been significant progress worldwide in the management of nuclear fuel's spent radioactive waste (HLW). Several countries have made important strides in advancing their plans for the construction of deep geologic repositories (DGRs) to safely dispose of their nuclear waste.

Finland led the way, with its nuclear waste management organization, Posiva Oy, submitting an application for an operating license for a DGR for spent fuel generated by the nuclear power plants of its owners. The facility, ONKALO, will be located on the island of Olkiluoto and is expected to begin final disposal in the mid-2020s. Sweden also approved SKB's application to build a DGR in Forsmark, and an encapsulation plant next to the Clab interim storage facility.

In Switzerland, Nagra selected Nordic Lagern as the site for the Swiss DGR, and is preparing the general license applications for the required facilities. Meanwhile, Canada's Nuclear Waste Management Organization (NWMO) narrowed down the possible locations for its DGR to two, and expects to name its preferred site by fall 2024.

The UK established four Community Partnerships to participate in the siting process for a DGR, with Nuclear Waste Services (NWS) responsible for identifying a site. Andra, the French organization responsible for managing all French radioactive waste, is expected to submit an application by the end of the year for a DGR in France that will contain HLW resulting from reprocessing of spent fuel assemblies from French nuclear power plants, as well as intermediate-level waste.

Overall, the progress made by these countries represents a tangible and sustainable step forward in the management of spent fuel and HLW, and brings us closer to the safe and effective long-term disposal of nuclear waste.

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Keywords: Spent fuel, HLW, DGR

Comparative Analysis of Burnup Uncertainty for Spent Fuel

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After spent fuel is stored in a dry storage container, it becomes difficult to obtain information on the fuel's characteristics. As a result, it is necessary to identify the characteristics of spent nuclear fuel in advance and secure the information necessary to establish delivery acceptance requirements for interim storage and disposal in the future. Therefore, it is necessary to evaluate the characteristics of spent fuel before loading dry storage casks.

In order to prepare for the dry storage of spent fuel, information on the basic characteristics of the fuel is required. As part of this information, it is also necessary to establish calculation criteria for spent fuel burnup.

Spent fuel burnup can be classified into three categories. The first is burnup evaluated using design codes (design burnup), the second is burnup measured by furnace instruments during power plant operation (actual burnup), and the third is burnup measured through measurement equipment (measured burnup).

This paper describes a comparative evaluation of design burnup, actual burnup, and measured burnup for specific fuels (40 bundles).

Keywords: Spent fuel, Burnup measurement, Actual burnup, Design burnup, Uncertainty

Thermal Evaluation of Consolidation Storage Canister for Spent Nuclear Fuel

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There have been a variety of issues related to spent nuclear fuel in Korea recently. Most of the issues are related to intermediate storage and disposal of spent nuclear fuel. However, recently, various studies have been started in advanced nuclear countries such as the United States to reduce spent nuclear fuel, focusing on measures to reduce spent nuclear fuel.

In this study, a simple preliminary assessment of the thermal part was performed for the consolidation storage method which separates fuel rods from spent nuclear fuel and stores them. The preliminary thermal evaluation was analyzed separately for storing the spent fuel in fuel assembly state and separating the fuel rods and storing them. The consolidation storage method in separating the fuel rods was advantageous in terms of thermal conductivity. However, detailed evaluation should be performed considering heat transfer by convection and vessel shape when storing multiple fuel bundles simultaneously.

Keywords: Spent nuclear fuel, Consolidation storage, Thermal evaluation

Review of the Fabrication of Square-gridded Baskets for Dry Storage of Spent Fuel

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The dry storage of spent fuel has become an increasingly important issue in the field of nuclear energy. Square-gridded baskets have been widely used for the storage of spent fuel because of their superior heat transfer and structural integrity. In this paper, we review the fabrication process of square-gridded baskets for dry storage of spent fuel. The review includes the design considerations, material selection, manufacturing methods, and quality control measures. We also discuss the challenges and opportunities for further improvement in the fabrication of square-gridded baskets. The fabrication of square-gridded baskets is a critical process for the safe and reliable dry storage of spent fuel. The review of the fabrication process highlights the importance of design considerations, material selection, manufacturing methods, and quality control measures. Continued efforts to improve the fabrication process will help to ensure the safe and secure storage of spent fuel.

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Keywords: Square-gridded baskets, Material selection, Quality control, Manufacturing method

Development of Integrated Analyzing and Training Simulator for Spent Nuclear Fuel Pool, CAREPool

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The Comprehensive Analyzer of Real Estimation for spent fuel POOL (CAREPOOL) has been developed for evaluating the thermal safety of a spent nuclear fuel pool (SFP) during the normal and accident conditions. The management of spent nuclear fuel function provides a management tool for spent nuclear fuel in the SFP. The fuel assemblies both in SFP and reactor side can be shown graphically in the screen. The loading sequence into transfer cask can be checked respectively in the CAREPOOL.

A basic heat balance equation was used to estimate the SFP temperature using the heat load calculated in the previous step. The characteristics of typical SFPs and associated cooling systems at reactor sites in the Korea were applied. Accident simulation like station black out leading to loss of SFP cooling or inventory is possible. Emergency cooling water injection pipe installed subsequent to the events at Fukushima 2011 is also modeled in this system. The CAREPOOL provides four main functions- management of spent nuclear fuel, decay heat calculation by ORIGEN-S code, estimation of the time to boil/fuel uncovering by thermal-hydraulics calculations, fuel selection for periodic spent fuel transferring campaign. All of these are integrated into the GUI based CAREPOOL system. The CAREPOOL would be very beneficial to nuclear power plant operator and trainee who have responsibility for the SFP operation.

Keywords: CAREPOOL, Spent fuel, ORIGEN, Station black out, Simulator

Evaluation of Fretting Wear on Spent Nuclear Fuel Rods During Transport

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Flow-induced vibration can lead to fretting wear damage of fuel rods and spacer grids in nuclear reactors. Similarly, during the transport of spent nuclear fuel assemblies, continuous vibration and intermittent impact might also result in fretting wear due to dynamic interaction. Therefore, it is important to evaluate fuel rod damage due to fretting wear under such transport conditions. This study examines spent nuclear fuel rod specimens fabricated with hydride cladding tubes and simulated pellets, with hydrogen content ranging from 200 to 700 ppm and oxide film thickness ranging from 25 to 100 micrometers. Tests were conducted under a worst-case scenario, assuming continuous exposure to that condition during the expected transport time. Results indicate that the wear depth of all rod specimens occurred within the oxide film, suggesting a high resistance to fretting wear during transportation.

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Keywords: Spent nuclear fuel assembly, Fretting wear, Transportation

Status of ‘Demo SILO’ Development Project to Verify an Integrity of CANDU Spent Fuel

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CANDU Spent Fuel (CSF) dry storage system, SILO, has been operated from 1992 at Wolsung under 50 year operating license. As of 2023, this system has been operated for over 30 years and its licensed remaining operation time is less than 20 years. When it faces the final stage of operation, it has only two options; moving to a centralized away-from-reactor storage or extending its license at-reactor. These two options have an inevitable common duty of confirming the CSF integrity by a “demonstration test”.

Since the degradation of CSF and structural materials in the SILO are critically dependent on temperature, two important goals of the ‘DEMO test’ were set as follows.

1. Design of ‘DEMO SILO’: Development of internal monitoring technology by transforming SILO design.
2. Accurate measurement and evaluation of the three-dimensional temperature distribution in the ‘DEMO SILO’

Based on operating real commercial SILO dimension, a conceptual “DEMO SILO” design has been developed from 2022. Because, unlike with commercial Silo, ‘Demo Silo’ must be disassembled and assembled, and have penetration holes. Safety evaluation technologies like structural, thermal and radiation protection analysis also have been developed with design work.

‘Demo SILO’ should evaluate an accurate 3D temperature distribution with minimal number of thermocouples and penetration holes to avoid disruption of internal flow and temperature distribution. For this reason, a ‘Best Estimate Thermal-Hydraulics evaluation system for SILO’ is under development and it will be essential for ensuring temperature prediction accuracy. Construction of a full-scale test apparatus to validate this technology will begin in 2024. In order to supply power to many heaters and monitor temperature gradient inside of this apparatus, it has modular design concept by dividing its whole body to axial 9 sub-bodies which looks like a donut containing a basket at center position.

Keywords: CANDU spent fuel, SILO, Fuel integrity, Demo SILO, Temperature distribution

Preliminary Analysis of the Characteristics of PWR Spent Nuclear Fuel With Respect to Initial Enrichment, Burnup, and Cooling Time Using ORIGAMI in SCALE 6.2.4

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Spent nuclear fuel (SNF) characterization is important in terms of nuclear safety and safeguards. Regardless of whether SNF is waste or energy resource, the International Atomic Energy Agency (IAEA) Specific Safety Guide-15 states that the storage requirements of SNF comply with IAEA General Safety Requirement Part 5 (GSR Part 5) for predisposal management of radioactive waste. GSR Part 5 requires a classifying and characterizing of radioactive waste at various steps of predisposal management. Accordingly, SNF fuel should be stored/handled as accurately characterized in the storage stage before permanent disposal. Appropriate characterization methods must exist to meet the above requirements.

The characterization of SNF is basically performed through destructive analysis/non-destructive analysis in addition to the calculation based on the reactor operation history. Burnup, Initial enrichment, and Cooling time (BIC) are the primary identification targets for SNF fuel characterization, and the analysis mainly uses the correlation identified between the BIC set and the other SNF characteristics (e.g., Burnup - neutron emission rate) for characterizing. So further identification of the correlation among SNF characteristics will be the basis for proposing a new analysis method.

Therefore, we aimed to simulate a SNF assembly with varying burnup, initial enrichment, and cooling time, then correlate other SNF properties with BIC sets, and identify correlations available for SNF characterization. In this study, the 'CE 16×16' type assembly was simulated using the SCALE-ORIGAMI code by changing the BIC set, and decay heat, radiation emission characteristics, and nuclide inventory of the assembly were calculated. After that, it was analyzed how these characteristics change according to the change in the BIC set. This study is expected to be the basic data for proposing new method for characterizing the SNF assembly of PWR.

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Keywords: Spent nuclear fuel, Initial enrichment, Burnup, Cooling time, Spent fuel characteristics

A Study on Re-racking of Spent Fuel Pool in Kori Unit 2

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The current storage capacity of the spent nuclear fuel at the Kori unit 2 has reached approximately 94% saturation, excluding emergency core capacity. As South Korea has not yet constructed any interim storage facilities to store spent nuclear fuel, one of possible options is to install high density storage racks in spent fuel pool at the reactor site to increase its capacity. The high density storage rack is a technology developed to allow the storage to have more spent nuclear fuel than originally planned, while still ensuring safety. It achieves this by reducing the storage gap between the spent nuclear fuel. The purpose of this study is to investigate an appropriate storage capacity for spent fuel pool in the Kori unit 2 and the factors to be considered during the high density storage rack installation.

Given that the Kori unit 2 is planning continued operation and Korea Hydro & Nuclear Power (KHNP) is preparing to install a temporary storage facility for spent nuclear fuel at the Kori nuclear site, it is reasonable to consider the installation of high density storage racks in the spent fuel pool as a storage solution for these issues. When evaluating the capacity of the spent fuel pool, the amount of spent nuclear fuel generated by other reactors in Kori nuclear site and the amount of spent nuclear fuel generated by continued operation of the Kori unit 2 should be taken into account. This study aims to consider these factors and evaluate the capacity of the spent fuel pool.

Furthermore, when installing high density storage rack for the spent nuclear fuel, it should be noted that the existing storage racks at the Kori unit 2 are welded to the liner plate, which may require additional cutting work. Therefore, it is necessary to review the suitable method for the cutting work. Additionally, assuming that divers need to access the area near the storage racks or cutting & welding devices require radiation protection in the area, it is essential to analyze the expected radiation level with computational code and propose appropriate measures to limit work time or establish a work zone.

Thus, this study evaluates appropriate capacity of spent fuel pool and work methods for the installation of high density storage rack in the spent fuel pool at the Kori unit 2. It is expected that this paper contributes to install high density storage racks in SFP safely.

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Keywords: Spent nuclear fuel (SNF), Spent fuel pool (SFP), Kori-2, SFP capacity, High density storage rack, SFP installation

Analysis of Trace Aluminium Oxide in UO_2 Using ICP-AES

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To improve the safety of nuclear fuel, research on the advanced nuclear fuel (UO_2) by adding various trace elements is being conducted. For example, the addition of metals such as Mo, Cr can improve the thermal conductivity of nuclear fuel, minimizing the diffusion of fission products. Trace metal oxide additives (SiO_2 , Cr_2O_3 , Al_2O_3 , etc.) can suppress the release of fission gases. In general, complete dissolution of the fuel sample is required for chemical analysis to determine its elemental compositions. Among widely used metal oxide additives, aluminum oxide is difficult to dissolve in nitric acid due to its excellent thermal and chemical stability.

In this study, we investigated on different chemical dissolution methods by applying a microwave digestion system under various acid solutions. We confirmed the validity of the digestion method by carrying out trace element analysis using an Inductively-Coupled Plasma Atomic Emission Spectrometer (ICP-AES).

Keywords: Aluminium oxide, Uranium oxide, Pretreatment, ICP-AES

Evaluation of Volatile Fission Products Inventory in the Spent Fuel Canning Container During the Drying Process

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In case of damaged spent fuels, it would require additional treatment for their transportation and storage to capture the radioactive fission products in a defined space. The canning container for the damaged spent fuels is one way to seal the radioactive fission products inside the container. In the Post Irradiation Examination Facility (PIEF) of KAERI, the Quiver container has been introduced for canning damaged spent fuels from Westinghouse Sweden. The main container body has been manufactured for particle-tightness of spent fuel. In addition, drying equipment is being prepared for gas-tightness of spent fuel. The drying equipment can remove water and fill the inert gas inside the container. Before drying inside the container, we evaluated the volatile fission products inventory because volatile fission products could be released during the drying process. Despite assuming highly conservative hypotheses for the inventory remaining in damaged fuel rods, the amount that could be released during the drying process was less and dose rate levels around the evacuation piping system were low.

Keywords: Spent fuel canning, Canning container, Drying process, Fission product, Fission gas

An Experimental Study on the Heat Flow of Tube Furnace by Air Flow in Tube

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In this paper, a basic study was conducted to observe the temperature inside the tube according to the heating temperature of the tube furnace. In a tube furnace, a tube is inserted, and the air space outside the tube is heated to increase the temperature of the gas inside the tube through conduction of the tube. Tube furnaces are widely used in research to capture volatile nuclides. In this case, a volatile nuclide capturing filter is inserted inside the tube, and an appropriate temperature is required to capture it. Since the tube furnace heats the air space outside the tube to the target temperature, a difference from the temperature inside the tube occurs. In particular, if a flow of gas occurs inside the tube, a larger temperature difference may occur. In order to confirm this temperature difference, an experimental device was constructed, and basic data was produced through several experiments. The following studies were conducted to produce data. First, the temperature of the air layer of the heating unit and the temperature inside the tube were measured in real time in the absence of gas flow inside the tube. Second, the temperature of the air layer of the heating unit and the temperature inside the tube were measured in real time while air having a certain temperature was flowing inside the tube. As a result of the experiment, when there is no flow inside the tube, when the heating target temperature is low, the temperature inside the tube is significantly lower than the target temperature, and when the target temperature is high, the temperature inside the tube approaches the target temperature. It was found that when there is about 20°C air flow inside the tube, the temperature inside the tube is significantly lowered even if the heating target temperature is high. In the future, additional research on changing the temperature of the gas flowing inside the tube will be conducted, and the results of this study are expected to greatly contribute to the design of a tube furnace that captures volatile nuclides.

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Keywords: Tube furnace, Volatile nuclide, Capturing filter, Air flow, Thermal flow, Heating

Experimental Investigation of Mechanical Strength of Dry Cask Storage Concrete According to Chloride Ion Concentration

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In concrete structures exposed to chloride environments such as seashore structures, chloride ions penetrate into the concrete. Chlorine ions in concrete react with cement hydrates to form Friedel's salt and change the microstructure. Changes in the microstructure of concrete affect the mechanical performance, and the effect varies depending on the concentration of chloride ions that have penetrated. However, research on the mechanical performance of concrete by chloride ion penetration is lacking. In this study, the effect of chloride ion penetration on the mechanical performance of dry cask concrete exposed to the marine environment was investigated. The mixture proportion of self-compacting concrete is used to produce concrete specimens. CaCl_2 was used to add chlorine ions, and 0, 1, 2, and 4% of the binder in weight were added. To evaluate the mechanical performance of concrete, a compressive strength test, and a splitting tensile strength test were performed. The compressive strength test was conducted through displacement control to obtain a stress-strain curve, and the loading speed was set to $10 \mu\text{ε}/\text{sec}$, which is the speed of the quasi-static level. The splitting tensile strength test was performed according to KS F 2423. As a result of the experiment, the compressive strength increased when the chloride ion concentration was 1%, and the compressive strength decreased when the chlorine ion concentration was 4%. The effect of the chloride ion concentration on the peak strain was not shown. In order to present a stress-strain curve model according to the chloride ion concentration, the existing concrete compressive stress-strain models were reviewed, and it was confirmed that the experimental results could be simulated through the Popovics model.

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Keywords: Compressive strength, Dry cask concrete, Chloride ion, Stress-strain curve

Effect of Vertical Shaking on the Reduction of Uranium (IV, VI) Oxides for Damaged Spent Fuel

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The stabilization technology for the damaged spent fuel is being developed to process the damaged fuel into sound pellet suitable for dry re-fabrication. It requires several treatments including oxidative decladding followed by reduction treatment for oxidized powder closely related to the quality of oxidized powders for pellet fabrication. For the development of operating condition for the reduction treatment, in this study, we evaluated the effect of air-cylinder based vertical shaking previously applied to oxidative decladding on powder reduction. For U_3O_8 of 50-100 g, the reduction test were applied with and without vertical shaking at 700°C under reduction atmosphere ($Ar + 4\%H_2$) and the concentration of hydrogen in effluent was measured to evaluate the reduction reaction. It was found that the vertical shaking system has allowed the reaction time of 50 g and 100 g U_3O_8 reduced by 33% compared to the test in static mode. Based on XRD analysis, the better crystallinity of the products was also achieved.

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Keywords: Fuel material, Reduction, Vertical shaking, Vibration

Economic Analysis of the Application of SFPI During Nuclear Decommissioning

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Spent Fuel Pool Island (SFPI) is a spent nuclear fuel storage pool that operates independently of existing nuclear facilities to safely manage SNF and minimize maintenance costs during the nuclear decommissioning process. Since the radiation controlled area can be dismantled before transporting SNF to a dry storage facility, the overall decommissioning period can be shortened, and the risk of occupational exposure during dismantling is reduced. In the US, various nuclear power plants have introduced SFPI for this reason.

In this paper, to analyze the economic feasibility of application of SFPI to nuclear power plants to be decommissioned, several scenarios are established in consideration of the decommissioning plan and schedule, SFPI and dry storage facility application schedule. Cost and benefit list (SFPI application cost, SNF management cost, SNF dry storage cask cost, etc.) for each alternative were derived, and economic analysis was conducted by applying the Net Present Value (NPV). As a result of the analysis, it is found that the application of SFPI during decommissioning is economically effective as the NPV showed a positive number even when uncertainty was taken into account.

Keywords: Spent fuel pool island, Decommissioning, Spent fuel management, Economic analysis, NPV

Preliminary Data Analysis of Surrogate Spent Nuclear Fuel in Shake Table Tests to Simulate Normal Conditions of Road and Sea Transport

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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.

The shake table test was performed with the input load produced conservatively from the data obtained from the road and sea transportation tests. The test input was produced based on the power spectral densities and shock response spectrums from the transportation tests. In addition to the test inputs from the road and sea tests, sine sweep input and half sine input were used to verify the vibration characteristics of assemblies under boundary conditions during normal conditions of transport.

Because the input load of the shake table test was produced conservatively, a slightly larger strain than the strain value measured in road and sea transportation tests was measured from the shake table tests. In the case of the sea test, it is considered that the process of enveloping the data in the 20 to 80 Hz range generated by the engine propeller system was performed excessively conservatively.

As a result of analyzing the test results for the difference in boundary conditions, it was confirmed that the test conditions of loading the basket generated a relatively large strain compared to the conditions of loading the disk assembly for the same input load. Therefore, it is concluded that a transportation cask having a structure in which a basket and a disk are separated, such as KORAD-21, is more advantageous in terms of vibration shock load characteristics under normal conditions of transport than a transportation cask having an integral internal structure in which a basket and a disk are a single unit. However, this effect will be insignificant because the load itself transmitted to the disk assembly is very small.

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Keywords: Normal conditions of transport, Spent nuclear fuel, Shake table test, Structural integrity

Preliminary Study of Chloride-Induced Stress Corrosion Cracking on the Welding Part of Canister Using DCPD Method

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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. In its simplest form it involves passing a constant current through the test piece and accurately measuring the electrical potential across the crack plane, and it is a suitable device to measure crack growth in real time.

The requirements for the CISCC simulation test selected based on the literature search results include test material 316 L, load range 1.75YS, positive displacement load, and 7% MgCl_2 concentration. In order to smoothly evaluate these various conditions, it was determined that it is advantageous to collect crack length data in real time using a DCPD device, rather than receiving and analyzing specimens maintained for a certain time in the chamber. Therefore, in this study, 4 types of test conditions in real time was built, and data collection on crack propagation could be performed in real time by using it.

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

Impact Analysis of Concrete Overpack of Dry Storage Cask

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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 accident conditions such as tip over and aircraft collision, both static material properties and dynamic properties are needed to evaluate the structural integrity of the concrete structures. Especially, it has been known to be difficult to estimate the resulted damage precisely where an aircraft collides with the degraded concrete structures at a high strain rate.

In this study, damage evaluation of concrete overpack due to aircraft collisions was conducted. First, in order to verify the impact analysis methodology, the aircraft impact analysis of plane concrete overpack was performed and compared with the test results previously conducted by our research team. Then, the impact analysis for the overpack of KORAD21C was performed. In the future, the radiation shielding analysis will be performed under the conditions to evaluate whether or not the radiation shielding ability is maintained.

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Keywords: Dry cask storage system, Interim storage facility, Aircraft collision, High strain rate

A Study of Mechanical Degradation Behavior of Spent Fuel in Long-term Dry Storage

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Research on the safety of nuclear spent fuel has been heavily experimented and modelled from a mechanical perspective. The issues of corrosion, irradiation creep, hydride and hydrogen embrittlement have been addressed more than two decades since the early 2000s. Among these degradation behavior, hydrogen embrittlement and hydride reorientation have been the most important topics for establishing the integrity of nuclear spent fuel and have been studied in depth.

In order to assess the safety of spent nuclear fuel, firstly, it is necessary to establish the safety criteria in all nuclear cycle, i.e., the failure criteria guidelines for nuclear fuel assemblies and nuclear fuel rods, and then examine the safety analysis. The contents of U.S.NRC Regulations, Title 10 General, Chapter 1 Code of Federal Regulation (CFR), Part 50, 71 and 72, describe the safety criteria for the safety assessment of nuclear fuel assemblies and nuclear fuel rods.

In this study, technically important points in safety analysis on nuclear fuel are checked through the reference of those NRC regulation. As result, we confirmed that the safety assessment of nuclear fuel after 20 years of interim storage is now being tested by ORNL and PNNL. There are not quantitative criteria related to material safety. However qualitative criteria which is dependent on environmentally condition describe the safety analysis. There is some literature study about DBTT, yield stress, ultimate tensile strength, flexural rigidity data. In FRAPCON code Modelling of yield strength and creep had been established, but radial hydride or hydride reorientation has not considered.

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

Attenuation Effect on the Effective Thermal Conductivity of Spent Fuel Assembly in Transport Cask

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This study investigates the behavior of the thermal conductivity among material properties in order to develop a thermal evaluation methodology of spent fuel assemblies in a transport cask. It is inefficient to model each element of the spent fuel assembly in detail, and it is generally calculated by modeling the effective thermal conductivity (ETC). The ETC model was developed to allow a much simpler representation of a spent fuel assembly within a fuel compartment by treating the entire spent fuel rod array and the surrounding fill gas within the confines of the compartment as a homogenous solid material. The fuel rod assembly and surrounding gas are modeled with an effective conductivity that is designed to yield an overall conduction heat transfer rate that is equivalent to the combined effect of local conduction and radiation heat transfer in a plane through the assembly.

When this model is applied to the transport cask, it tends to predict the cladding peak temperature lower than the results of detailed model in which the fuel rod arrangement and shape of the fuel assembly are simulated. As for the tendency of the error, the model tended to under-predict when basket temperature was lower than a certain temperature, and over-predict when it was higher.

The purpose of this study is to investigate the attenuation effect of the cladding peak temperature on the related variables when the ETC model is applied to the transport cask. In addition, based on the thermal characteristics of this model, a correction factor that can compensate for this attenuation effect is presented. This correction factor is obtained by finding the difference between a separate ETC homogeneous model and a separate detailed fuel model, rather than directly applying the ETC calculated from the detailed fuel model to the transport cask.

Keywords: Attenuation effect, Effective thermal conductivity, Heat transfer, Spent fuel assembly, Spent fuel transport cask

Study on Methods for Obtaining Reliability in the Manufacture of Metal Cask for Transportation and Storage of Spent Nuclear Fuel

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As regulations on carbon emissions increase, the interest in renewable energy is also increasing. However, the efficiency of renewable energy generation is highly low and has limitations in replacing existing energy consumption. In terms of this view, nuclear power generation is highlighted because it has the advantage of not emitting carbon. And accordingly, the amount of spent nuclear fuel is going to increase naturally in the future. Therefore, it will be important to obtain the reliability of containers for transporting safely and storing spent nuclear fuel.

In this study, a method for verifying the integrity and airtightness of a metal cask for the safe transportation and storage of spent nuclear fuel was studied. Non-destructive testing, thermal stability, leakage stability, and neutron shielding were demonstrated, and as a result, suitable quality for loading spent nuclear fuel could be obtained. Furthermore, it is meaningful in that it has secured manufacturing technology that can be directly applied to industrial field by verifying actual products.

Keywords: Spent nuclear fuel, Metal cask, Air tightness, Neutron shielding

Radioactivity Calculations and Benchmarking of Nuclides in Spent Fuel

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In this study, radioactivity of Cs-134, Cs-137, and Eu-154, which are gamma-emitting nuclides among fission products of spent fuel, was analyzed as a tool to measure the burnup of spent fuel nondestructively. This nuclide has a unique gamma-ray energy, allowing the amount of the isotope to be estimated based on the intensity of the gamma-ray at a specific energy. The SCALE 6.2 ORIGAMI (ORIGen AsseMbly Isotopics) module and the latest ORIGEN-arp library were used for computational analysis. The spent fuel samples were selected as WH14×14 with an enrichment of 1.5~5.0wt%, a burnup of 10~60 GWD/MTU, and a cooling time of 0~40 years.

The analysis results were benchmarked using SFCOMPO experimental data provided by OECD/NEA, including isotope inventory and uncertainty measured by destructive radiochemical analysis, fuel assembly design data required for benchmark model development, reactor design information, and operating history information. 16 similar spent fuels were selected from SFCOMPO data and the calculation results of Cs-134, Cs-137, and Eu-154 were compared. The average error of the Cs-134 radioactivity calculation result was 2.81%, and the maximum error was 6.70%. The average errors of Cs-137 and Eu-154 were 2.42% and 4.95%, respectively, and the maximum errors were 5.40% and 14.91%, respectively.

Keywords: Spent fuel, Radioactivity calculations, ORIGAMI, SFCOMPO

Concept of the Precipitation Reactor Design for Reduction of Molten Salt Amount Transferred to the Distillation Unit

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A phosphorylation (phosphate precipitation) technology of metal chlorides is considering as a proper treatment method for recovering the fission products in a spent molten salt. In KAERI's previous precipitation tests, the powder of lithium phosphate (Li_3PO_4) as a precipitation agent reacted with metal chlorides in a simulated LiCl-KCl molten salt. The reaction of metal chlorides containing actinides such as uranium and rare earths with lithium phosphate in a molten salt was known as solid-liquid reaction. In order to increase the precipitation reaction rate the powder of lithium phosphate dispersed by stirring thoroughly in a molten salt. As one of the recovery methods of the metal phosphates precipitated on the bottom of the molten salt vessel cutting method at the lower part of the salt ingot is considered. On the other hand, a vacuum distillation method of all the molten salt containing the metal phosphates precipitates was proposed as another recovering method. In recent study, a new method for collecting the phosphorylation reaction products into a small recovering vessel was investigated resulting in some test data by using the lithium phosphate ingot in a molten salt containing uranium and three rare earth elements (Nd, Ce, and La). The phosphorylation experiments using lithium phosphate ingots carried out to collect the metal phosphate precipitates and the test result of this new method was feasible. However, the reaction rate of test using lithium phosphate ingot is very slower than that of test using lithium phosphate powder. In this presentation, the precipitation reactor design used for phosphorylation reaction shows that the amount of molten salt transferred to the distillation unit will reduce by collecting all of the metal phosphates that will be generated using lithium phosphate powder into a small recovering vessel.

Keywords: Uranium, Rare earths, Metal phosphate, Molten salt, Precipitation reactor design

Comparison of Results According to the Number of Measured Faces in Burnup Measurement of Spent Fuel

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The burnup of spent fuel is one of the important management items that must be managed before storing the fuel in dry storage facilities, as well as for transportation and disposal in the future. Currently, the burnup of spent fuel is managed by calculating the design burnup at the time of design and measuring the real burnup using in-reactor measurement devices.

Furthermore, to ensure the reliability of such data, the burnup of spent fuel can be measured using burnup measurement equipment to compare and analyze the data. In fact, KHNP is measuring the burnup of spent fuel using the burnup measurement equipment (SICOM-NG-FA) developed by ENUSA in Spain.

The burnup measurement equipment analyzes the axial burnup profile of spent fuel using gamma and neutron detectors. Burnup measurement is performed by moving the spent fuel up and down inside the measurement equipment and measuring the burnup of the fuel surface facing the gamma and neutron detectors.

This paper aims to compare the results of measuring the burnup of spent fuel on two sides versus four sides using the burnup measurement equipment.

Keywords: Spent fuel, Burnup, Detector

Evaluation of Thermal and Leaching Properties of Ceramic Wastes for Immobilization of Group II Nuclides

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Heat-generating nuclides such as Cs-137 and Sr-90 should be separated from spent nuclear fuel to reduce the short-term thermal load on the repository facility. In particular, Sr-90 must be separated because its decay process generates high temperatures. Recently, the Korea Atomic Energy Research Institute (KAERI) has been developing a waste burden minimization technology to reduce the environmental burden resulting from the disposal of spent nuclear fuel and maximize the utilization of the disposal facility. The technology incorporates a nuclide management process that maximizes disposal efficiency by selectively separating and accumulating key nuclides from spent nuclear fuel, such as Cs, Sr, I, TRU/RE, and Tc/Se. Sr nuclides dissolve in the chloride phase during the chlorination process of spent nuclear fuel and are recovered as carbonate or oxide through reactive distillation or reactive crystallization. Due to their chemical similarity, Ba nuclides are recovered along with Sr nuclides during this process. In this study, we prepared a ceramic waste form for group II nuclides, $\text{Ba}_{(x)}\text{Sr}_{(1-x)}\text{TiO}_3$ ($x=0, 0.25, 0.5, 0.75, 1$), using the solid-state reaction method, taking into account the different ratios of Sr/Ba nuclides produced during the nuclide management process. Regardless of the Sr/Ba ratio, the established waste form fabrication process was able to produce a stable waste form. Physicochemical properties, including leaching and thermal properties, were evaluated to determine the stability of group II waste forms. In addition, the radiological properties of waste forms of $\text{Ba}_{(x)}\text{Sr}_{(1-x)}\text{TiO}_3$ with varying Sr/Ba ratios were evaluated. These results provided fundamental data for the long-term storage and management of waste forms containing group II nuclides.

Keywords: Strontium-90, Group II nuclides, Ceramic waste form, Centerline temperature, Nuclide management process

Segregation Method of the Recovered Actinide Metal Mixture From a Molten Salt Medium in the Back-end Fuel Cycles System

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When the recycling technology of spent nuclear fuels (SNF) for future nuclear reactor systems and the treatment technology of SNF for disposing of in a disposal site use a molten salt such as LiCl-KCl eutectic as a processing medium one of the essential unit processes is a distillation process that remove the salt component mixed with fission products recovered. Especially, in case of Pyro-SFR recycling system the recovered nuclear fuel materials such as U, TRU and some of rare earths come from main three processes (electro-refining, electro-winning, and drawdown processes) for recycling of SNF. These recovered fuel materials contain large portion of molten salt or liquid cadmium which requires removal of them by distillation. In spent nuclear fuels discharged from PWR the portion of composing element is as follows. Uranium is about 95%, other actinides such as transuranic elements (TRU; Np, Pu, Am, Cm) is about 1%, the rare earths (lanthanides) is about 1%, and the other elements is about 3%. For example, americium (Am) in the recovered fuel materials has a problem that the reported loss of Am inevitably occurs during the vacuum salt distillation operation. A new segregation method of AMM (actinide metal mixture)-salt system is based on the difference in melting point of the actinide elements. It is possible to apply this segregation method to recovering other actinides from AMM with accompanied salt because of relatively large amount and lower melting point of a specific element in other actinides avoiding vacuum salt distillation. This new segregation method successfully tested using a surrogate element such as aluminum due to its similar melting point with a specific element. The segregation principle is solid-liquid separation, thus the solidified actinides mixture ingot can take out of a molten salt medium.

Keywords: Spent nuclear fuel, Actinides, Uranium, Rare earths, Molten salt, Segregation

Sensitivity Analysis of Waste Properties by Performance Change in Unit Process of Radionuclide Management Process

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Korea Atomic Energy Research Institute is developing a radionuclide management processes as a conditioning technology to reduce the burden of spent fuel disposal. The radionuclide management process refers to a process managing radionuclides with similar properties by introducing various technology options that can separate and recover radionuclides from spent fuels. In particular, it is a process aimed at increasing disposal efficiency by managing high-heat, high-mobility, and high-toxic radionuclides that can greatly affect the performance of the disposal system.

Since the radionuclide management process seeks to consider various technology options for each unit process, it may have several process flows rather than have a single process flow. Describing the various process flows as a single flow network model is called the superstructure model. In this study, we intend to develop a superstructure model for the radionuclide management process and use it as a model to select the optimal process flow. To find the optimal process flow, an objective function must be defined, and at the fuel cycle system level multiple objectives such as effectiveness (disposal area), safety (explosure dose), and economics (cost) can be considered. Before performing the system-level optimization, it is necessary to select candidates of process flow in consideration of waste properties and process efficiency at the process level.

In this study, a sensitivity analysis is conducted to analyze changes in waste properties such as decay heat and radioactivity when the separation ratio varies due to the performance change for each unit process of the radionuclide management process. Through this analysis, it is possible to derive a performance range that can have waste properties suitable for following waste treatment, especially waste form manufacturing. It is also possible to analyze the effect of waste properties that vary according to the performance change on waste storage and management approaches.

Keywords: Radionuclide management process, Sensitivity analysis, Waste property

Fatigue Characteristics of Spent Nuclear Fuel Rod During NCT

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Korean MMTT project has been launched in order to clarify the vibration and shock loads under normal condition and transportation (NCT) in Korean geological and transportation conditions and to evaluate the integrity of SNF under such a transportation load. To evaluate the integrity of the SNF during normal land and sea transport tests, a representative SNF that represents the entirety of the different types of SNFs stored in the spent fuel pool of the power plant should be selected. And, it is necessary to make the test assembly to have a statically and dynamically similar behavior with the actual SNF. Therefore, in this project, we selected two types of fuel assembly that are expected to exhibit relatively conservative behavior under NCT, and these assemblies are being fabricated into surrogate test assemblies to have a similar characteristic as actual SNF based on the accumulated data from the poolside examination and the hot cell test so far. Tests were conducted for NCT conditions.

In addition, a fatigue test was performed to integrity of the nuclear fuel rods under NCT conditions. Nuclear fuel assemblies are transported while being laid inside the cask under NCT, and are exposed to external shocks and vibrations. At this time, the fuel rod between the grid and grid is exposed to bending motion by this external force. For this simulation, a fixture was developed and used for static bending tests and bending fatigue tests. To simulate spent nuclear fuel rod specimens, hydrogen reorientation Zry-4 cladding was used and simulated pellets made of stainless steel were applied. And also, it was bonded using epoxy to give bonding conditions between the inside and the pellet.

As a result of the test, cracks occurred due to the concentrated load between the pellets, resulting in damage to the fuel rod. The fatigue results showed a similar trend compared to the results performed by ORNL, and the lower bound fatigue curve presented by NUREG-2224 was also satisfactory.

Acknowledgements

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Keywords: Spent nuclear fuel assembly, Fatigue, Reversible, Transportation, KMMTT

Development of Cutting Machine for Post-irradiation Examination of Spent Fuel Rod in Hot Cell

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Korea Atomic Energy Research Institute (KAERI) has been operating the Post Irradiation Examination Facility (PIEF) for spent fuel. The facility has pools and hot cells for handling and examining fuel assemblies and rods. In the first hot cell, non-destructive tests such as visual inspection, defect detection, oxide layer thickness measurement, and gamma scanning are performed on a full-length fuel rod. Then, the fuel rod is transported to the next hot cell for measuring the rod internal pressure (RIP). After the RIP measurement, the fuel rod is cut by a cutting machine to make samples for destructive tests. Currently, the existing cutting machine is broken, so a new machine needed to be designed and manufactured. The major considerations for designing the cutting machine were convenience of remote handling and decontamination. The machine was modularized and its handling parts were designed to be easily controlled by manipulators. The cover was designed to prevent radioactive contamination of the surrounding area.

Keywords: Hot cell, Fuel rod, Cutting machine, Post irradiation examination, Fuel cladding

Development of Neutron Shielding Resin Composition Applied to Spent Nuclear Fuel Dry Cask

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A radiation shielding resin with thermal stability and high radiation shielding effect has been developed for the neutron shielding resin filled in the shielding shell of dry storage/transport cask for spent nuclear fuel. Among the most commercially available neutron shielding resins, epoxy and aluminum hydroxide boron carbide are used. But in case of the resin, hydrogen content enhances the neutron shielding effect through optimization of aluminum hydroxide, zinc borate, boron carbide, and flame retardant. We developed a radiation shielding material that can increase the boron content and have thermal stability. Flame retardancy was evaluated for thermal stability, and neutron shielding evaluation was conducted in a research reactor to prove the shielding effect. As a result of the UL94 vertical burning test, a grade of V-0 was received. Therefore, it was confirmed that it had flame retardancy.

According to an experiment to measure the shielding rate of the resin against neutron rays using NRF (Neutron Radiography Facility), a shielding rate of 91.54% was confirmed for the existing resin composition and a shielding rate of 96.30% for the developed resin composition. A 40 M SANS (40 M Small Angle Neutron Scattering Instrument) neutron shielding rate test was performed. Assuming aging conditions (6 hours, 180 degrees), the shielding rate was analyzed after heating. As a result of the experiment, the developed products with 99.8740% and 99.9644% showed the same or higher performance.

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Keywords: Neutron shielding, Flame retardancy, Cask, Epoxy resin, Boron carbide, Zinc borate

Analysis and Design of Rod Extractor and Cutter for Spent Fuel Assembly Dismantling in Head-End

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An important goal of dismantling process is the disassembling of a spent nuclear fuel assembly for the subsequent extraction process. In order to design the rod extractor and cutter, the major requirements were considered, and the modularization design was carried out considering remote operation and maintenance. In order to design the rod extractor and cutter, these systems were analyzed and designed, also the concept on the rod extraction and cutting were considered by using the solid works tool. The main module consists of five sub-modules, and the function of each is as follows. The clamping module is an assembly fixing module using a cylinder so that the nuclear fuel assembly can be fixed after being placed. The Pusher module pushes the fuel rods by 2 inches out of the assembly to grip the fuel rods. The extraction module extracts the fuel rods of the nuclear fuel assembly and moves them to the consolidation module. The consolidation module collects and consolidates the extracted fuel rods before moving them to the cutting device. And the support module is a base platform on which the modules of the main device can be placed. The modules of level 2 can be disassembled or assembled freely without mutual interference. For the design of fuel rods cutter, the following main requirements were considered. The fuel rod cut section should not be deformed for subsequent processing, and the horizontally mounted fuel rods must be cut at regular intervals. The cutter should have the provision for aligning with the fuel rod, and the feeder and transport clamp should be designed to transfer the fuel rods to the cutting area. The main module consists of 6 sub-modules, and function of each is as follows. The cutting module is a device that cuts the fuel rods to the appropriate depth for notching. The impacting module is a device that impacts the fuel rods and moves them to the collection module. The transfer module is a device that moves the fuel rods to the cutting module when the aligned fuel rods enter the clamp module. The clamping module is a device to clamp the fuel rods before moving them to the cutting module. The collection module is a container where the rod-cuts are collected, and the support module is a base platform on which the modules of the main device can be placed. The module of level 3 can be disassembled or assembled after the cutting module of level 2 is installed, and the modules of level 2 can be disassembled or assembled freely without mutual interference.

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Keywords: Design, Extractor, Cutter, Spent, Fuel, Dismantling

Considerations for the Ventilation Control Analysis Using Fire Modelling Tools on the SNF Dry Storage Facility

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South Korea has been storing UNF in spent fuel pool dry storage facility within Nuclear Power Plants. The dry storage facility of used nuclear fuel (UNF) is essential to sustain safety and sustain stable operation of a nuclear power plant. Most abroad countries have attempted to develop a variety of dry storage facility for used nuclear fuel in order to retain the safe restoration. Many studies have been conducting to safety evaluation for the dry storage facility. However, there is not a ventilation evaluation in the wake of fire event that could influence of the thermal effect on the dry storage facility, even though it will likely to occur fire events such as wildfire, air craft crash. In practice, it happened to catastrophic disaster due to the wild fire adjacent to ul-jin mountain. Also, it happened to fire accident near to the Japonia NPP in Ukraine territory caused of military air plane missile. It has not mostly been studied on the ventilation evaluation considered to thermal safety in the dry storage facility excepted for some researches. It could need the mechanical ventilation systems such as HVAC system in the dry storage system, so that thermal effect can be reduced. In this study, we conducted to the ventilation control modelling by using fire modelling tool (Fire Dynamic Simulator v.6.7). The ventilation scenarios made up for 3 case that can compare flowrate variation with ventilation control.

As a result of modelling, there is no differentiation between ventilation control using performance curve with not using performance curve even though the pressure fluctuation would be increased, compared with the case of considering performance curve. Second, it evaluated that the mode for fraction control would occur to pressure rise in the state of controlling the ventilation system flowrate. However, sensitivity of flowrate control was more decreased below less than 5 seconds. Third, in the case of on/off control system revealed more higher resolution than other cases caused by flowrate variation.

These results could be considered as the design guidelines for the development dry storage facility to improve the thermal performance that can reduce thermal risk. Furthermore, the study results would expect HVAC system installed in dry storage to help automatic ventilation control relevant to dry storage safety increased.

Keywords: Spent nuclear fuel, HVAC ventilation analysis, Dry storage facility, Flowrate control

Platinum Group Metals Recovery From Highly Radioactive Raffinate of NPP SNF Reprocessing

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Precipitation of platinum group metals (Rh, Ru, Pd, so-called MPG) from the melt essentially affects the reliability of installations for vitrification of high-level liquid radioactive waste (HLW). To date, it is difficult to find an approach which allows simultaneous recovery of all three metals.

The aim of our work was to select a sorbent that would provide simultaneous up to complete recovery of given metals. The following inorganic materials were tested as sorbents – yellow blood salt (YBS). and hexacyanoferrates of iron, aluminum, copper and nickel. The degree of metal recovery was studied is influenced by the temperature and concentration of nitric acid.

Only palladium was completely recovered using YBS. At the same time, specially prepared iron hexacyanoferrate (HCF-Fe) under optimal experimental conditions recovers almost all Pd and more than 95% and 90% of Rh and Ru, respectively.

The behavior of fission products, including the main dose-forming components of HLW (Cs, Sr) and Mo, U, Ag, REE) in the course of MPG recovery was studied. The experiments were carried using both multicomponent model solutions and real raffinates. Options for further management of the recovered metals have been worked out.

Thus, the proposed method of metal recovery seems promising for the development of a technology for the removal of MPG from nitric HLW during the reprocessing of the spent nuclear fuel (SNF) before vitrification. The recovered metals can be probably used in various technological processes. Also, this method can provide the MPG recovery from low-concentration tail solutions.

Keywords: Ruthenium, Rhodium, Palladium, Highly active waste

3분과

고준위폐기물처분(Oral)

Alkali Borosilicate Glass for Vitrification of Molybdenum-rich Nuclear Waste

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Vitrification is one of the best ways to immobilize high-level radioactive waste (HLW) worldwide over the past 50 years. Since the glass matrix has a medium (3.0-5.5 Å) and short (1.5-3.0 Å) periodicity, it can accommodate most elements from the periodic table. Borosilicate glass is the most suitable glass matrix for vitrification due to its high chemical durability, high waste-loading capacity, and good radiation resistance.

Mo is a fission product contained in liquid waste generated in the process of reprocessing spent nuclear fuel and exists in the form of MoO_4^{2-} in the glass. MoO_4^{2-} forms a depolymerization region without directly connecting with the glass network former. When the concentration of Mo increases in the depolymerization region, it combines with nearby alkali or alkaline earth cations to form a crystalline molybdate phase. Phase separation and crystallization in the glass can degrade the performance of the material because it changes the physical and chemical properties of the glass. In particular, since alkali molybdate has high water solubility when it forms crystals containing radioactive elements such as Cs, there is a risk of leakage of radionuclides by groundwater during deep underground disposal.

Therefore, in this study, the most stable glass-ceramic composition was developed using various alkali elements, and the difference in phase separation and crystallization behavior in glass and the stability of the solidified body were analyzed by structural analysis of the glass network and alkali molybdate. The cause of the difference in crystallization of alkali molybdate according to the type of alkali cation is structurally analyzed, and using this, research is conducted to increase the Mo content in the glass without crystallization.

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Keywords: Nuclear waste glass, Borosilicate glass, Molybdenum, Crystallization, Glass structure

Study on the Change in Neutron Absorbency Due to BSS Corrosion in the Accelerated Environment in the Spent Nuclear Fuel Pool

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In Korea, borated stainless steel (BSS) is used as a storage rack in spent fuel pools (SFP) to maintain the nuclear criticality of spent fuels. As the number of nuclear power plants and the corresponding amount of spent fuels increased, the density in SFP storage rack also increased. In this regard, maintaining subcriticality of spent nuclear fuels became an issue and BSS was selected as the structural material and neutron absorber for high density storage rack. Since it is difficult to replace the storage rack, corrosion resistance and neutron absorbency are required for long period.

BSS is based on stainless steel 304 and is specified in the ASTM A887-89 standard depending on the boron concentration from 304B (0.20-0.29% B) to 304B7 (1.75-2.25% B). Due to the low solubility of boron in austenitic stainless steel, metallic borides such as (Fe, Cr) 2B are formed as a secondary phase. Metallic borides could cause Cr depletion near it, which could decrease the corrosion resistance of the material.

In this paper, the long-term corrosion behavior of BSS and its oxide microstructures are investigated through accelerated corrosion experiment in simulated SFP conditions. Because the corrosion rate of austenitic stainless steel is known to be dependent on the Arrhenius equation, a function of temperature, the corrosion experiment is conducted by increasing the experimental temperature. Detail microstructural analysis is conducted using a scanning electron microscope, transmission electron microscope and energy dispersive spectrometer.

After oxidation, a hematite structure oxide film is formed, and pitting corrosion occurs on the surface of specimens. Most of the pitting corrosion is found at the substrate surface because the corrosion resistance of the substrate, which has low Cr content, is relatively low. Also, the oxidation reaction of B in the secondary phase has the lowest Gibbs free energy compared to other elements. Furthermore, oxidation of Cr has low Gibbs free energy, which means that oxidation of B and Cr could be faster than other elements. Thus, the long-term corrosion might affect the boron content and the neutron absorption ability of the material.

Using boron's high cross-section for neutrons, the neutron absorption performance of BSS was evaluated through neutron transmission tests. The effect of the corrosion behavior of BSS on its neutron absorption performance was investigated. Samples simulated to undergo up to 60 years of degradation before corrosion through accelerated corrosion testing did not show significant changes in the neutron shielding ability before and after corrosion. This can be explained in relation to the corrosion behavior of BSS. Boron was only leached out from the secondary phase exposed on the surface, and this oxidized secondary phase corresponds to about 0.17% of the volume of the total secondary phase. This can be seen as a very small proportion compared to the total boron content and is not expected to have a significant impact on neutron absorption performance.

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Keywords: Neutron absorber, Borated stainless steel, Long-term corrosion, Accelerated corrosion, Neutron absorbency

A Geochemical Study on the Uranium Mobility in a Deep Geological Repository: A Natural Analogue Approach Using a Uranium Deposit in the Ogcheon Metamorphic Belt

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The mobility of uranium (U) in the environment 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 mobility of uranium in a deep geological repository by a natural analogue approach using a uranium deposit in the Ogcheon Metamorphic Belt (OMB). Uranium contents of rock samples from the study site ranged from 1.3 to 71 ppm (average 17.4 ppm). Uranium minerals found in the study site were mostly uraninite ($\text{U}^{\text{IV}}\text{O}_{2+x}$) and uranothorite ($(\text{U}^{\text{IV}}, \text{Th})\text{SiO}_4$). The concentrations of U in the groundwater samples were very low (0.025~0.690 ppb) even though redox conditions are weakly oxidizing.

Calculation results for U speciation in groundwater samples showed that major dissolved uranium species in the groundwater samples are mainly as calcium uranyl (UO_2^{2+}) carbonate complexes such as $\text{Ca}_2\text{UO}_2(\text{CO}_3)_3(\text{aq})$ and $\text{CaUO}_2(\text{CO}_3)_3^{2-}$. However, the activity ratios between ^{234}U and ^{238}U ($\text{AR}(^{234}\text{U}/^{238}\text{U})$) showed U behavior in reducing conditions although the groundwater conditions were not reducing conditions and major dissolved U species were U(VI) species. Results from electron microscopic analyses for rock samples showed that major uranium minerals were U(IV) minerals such as uraninite and uranothorite. We could not identify other uranyl minerals and altered minerals from uraninite. This means that the geochemical condition of the study site has been maintained a reducing condition although the groundwater condition was a weakly oxidizing condition. Thus, the dissolution of uranium is strongly limited by the low solubility of uraninite. It is not obvious how the reducing condition of the study site has been maintained. Reducing agents such as pyrite, organic materials, and reducing bacteria might contribute to maintaining the reducing condition although further studies will be necessary.

Results from this study imply that uranium mobility will be greatly limited by low dissolution of uraninite into groundwater if the reducing condition is well reserved. This limited mobility of uranium will be also contributed by low possibility of uraninite alteration into uranyl minerals which have a higher solubility than uraninite.

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Keywords: Natural analogue study, Uranium deposit, Mobility of uranium, Black shale, Uranium speciation, Uranium mineral

Migration Behavior of Simulated HLW Leachate in Granite and Biotite Gneiss Aquifers

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The safe disposal of high-level radioactive waste is a critical concern in many countries, especially in the context of the increasing use of nuclear power to overcome climate change. To provide a comprehensive understanding of the behavior of the radionuclides in the crystalline natural barrier, sorption of the artificially synthesized high-level radioactive waste (HLW) leachate was conducted.

Granite (-1,000 m from ground level) and biotite gneiss (-100 m from ground level) rock cores were collected from Gyeongju and Gwacheon, respectively. The rock cores were milled with a jaw crusher and steel disk mill and then sieved. The crushed rocks with a diameter of 0.6 – 1.0 mm were selected, washed three times with deionized water, and then dried. To synthesize the simulated HLW leachate, representative elements (U(VI), Se(IV), Mo(VI), and Ni(II)) were added to natural groundwater collected from Gyeongju. The kinetic sorption experiment was performed in a polypropylene bottle with a solid-to-liquid ratio of 100 g/L in the orbital shaking incubator (200 rotations per min, 25.0°C). After the sorption, the supernatants were filtered by a 0.2-μm polytetrafluoroethylene syringe filter and subsequently analyzed by inductively coupled plasma-mass spectrometry (ICP-MS). Through the kinetic change of aqueous concentration, the contact time has been determined to be 7 days. Ni(II) showed the highest distribution coefficients ($K_d = 0.81 \text{ L/m}^2$ for granite and $8 - 16 \text{ L/m}^2$ for biotite gneiss), followed by U(VI) ($K_d = 0.03 - 0.04 \text{ L/m}^2$ for granite and $0.04 - 0.05 \text{ L/m}^2$ for biotite gneiss). Highly mobile nuclides such as Se(IV) ($K_d = 0.02 \text{ L/m}^2$ for granite and 0.03 L/m^2 for biotite gneiss) and Mo(VI) ($K_d = 0.01 - 0.02 \text{ L/m}^2$ for granite and 0.01 L/m^2 for biotite gneiss) showed the lowest distribution coefficient.

Our study provides insights into the migration-retention behaviors of the HLW leachate with granite and biotite gneiss in geological systems and verifies the sorption parameters, e.g., distribution coefficients, experimentally produced by other groups to ensure the safe disposal of HLW.

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Keywords: Migration, Sorption, Distribution coefficient, Granite, Biotite gneiss, Natural groundwater

Evaluation of Leaching and Thermal Properties of Silver Tellurite Glass for Iodine Immobilization

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Many countries have used nuclear power to generate electricity. Uranium-235, which is used as fuel in nuclear power plants, produces many fission products. Among them, iodine-129 is problematic due to its long half-life (1.57×10^7 years) and high diffusivity in the environment. If it is released into the environment without any treatment, it could have a major impact on humans and ecosystems. Therefore, it must be treated into a stable form through capture and solidification. Iodine can be captured in the form of AgI through silver-loaded zeolite filters in off-gas treatment processes. However, AgI could be decomposed in the reducing atmosphere of groundwater, so it must be converted into a stable form.

In this study, Al_2O_3 , Bi_2O_3 , PbO , V_2O_5 , MoO_3 , or WO_3 were added to the iodine solidification matrix, AgI-Ag₂O-TeO₂ glass. The glass precursors were mixed to the appropriate composition and placed in an alumina crucible. After heat treatment at 800°C for 1 hour, the melt was quenched in a carbon crucible. The leaching behavior and thermal properties of the glass samples were evaluated. The PCT-A test for leaching evaluation showed that the normalized releases of all elements were below 2 g/m², which satisfied the U.S. glass wastefrom leaching regulations. Differential scanning calorimetry (DSC) was performed to evaluate the thermal properties of all glass samples. The addition of MoO₃ or WO₃ to the AgI-Ag₂O-TeO₂ glass increased the glass transition temperature (T_g) and crystallization temperature (T_c) while maintaining the glass stability. The similar relative electro-static field values of MoO₃, and WO₃ which are approximately three times that of the glass network former TeO₂, could provide sufficient force to the TeO₂ interacting with the non-bridging oxygen forming Te-O-M (M=V, Mo, W) links. The high electrostatic forces of Mo and W increased the glass network cohesion and prevented the crystallization of the glass.

Keywords: Radioactive waste, I-129, Silver tellurite glass, Glass stability, Leaching property

The Impact of Gas-mediated Transport on Nuclide Behavior: Results From an Experimental Study

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The design of a radioactive waste disposal system should include both natural and engineered barriers to prevent radionuclide leakage and groundwater contamination. Colloids and gases can accelerate the movement of radionuclides and affect their behavior. It is important to consider these factors in the long-term stability evaluation of a deep geological repository. An experimental setup was designed to observe the acceleration of nuclide behavior caused by gas-mediated transport in a simulated high temperature and pressure environment, similar to a deep disposal repository. The study used specimens to simulate gas flow in engineered barriers, based on conditions 1000 years after repository closure. In the experiment, bentonite WRK with a dry density of 1.61 g/cm^3 was used after compaction. Measurements were taken of the saturation time and gas permeability of compacted bentonite. In this study, gas was injected into saturated buffer materials at various pressures to evaluate the penetration phenomenon of the buffer material according to the gas pressure. It was observed that gas penetrated the buffer material and moved upward in the form of gas bubbles at a specific pressure. Furthermore, when a flow was continuously induced to penetrate the buffer material, erosion occurred, and the eroded particles were found to be able to float upward or be transported by gas bubbles. In future studies, analysis will be conducted on the transport rate of fine particles according to the size of gas bubbles and the characteristics of the nuclides adsorbed on the fine particles.

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Keywords: Nuclide behavior, Gas migration, Accelerated gas behavior, Bentonite, Deep geological repository

Development of Scenarios Using a Hybrid Approach for Deep Geological Disposal of Spent Nuclear Fuel

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To obtain a license for a deep geological disposal repository for spent nuclear fuel, it is necessary to perform a safety assessment that quantifies the radiological impact on the environment and humans. One of the key steps in the safety assessment of a deep geological repository is the development of scenarios that describe how the repository evolves over the performance period and how events and processes affect performance. In the field of scenario development, demonstrating comprehensiveness is critical, which describes whether all factors that are expected to have a significant impact on the repository's performance have been considered. Mathematical proof of this is impossible. However, If the scenario development process is logical and systematic, it can support the claim that the scenario is comprehensive. Three primary approaches are being considered for scenario development: 'Bottom-up', 'Top-down', and 'Hybrid'. Hybrid approach provides a more systematic and structured process by considering both the FEPs (Features, Events, Processes) and safety functions utilized in the bottom-up and top-down approaches. Many countries that develop recent scenarios prefer demonstrating scenario comprehensiveness using a hybrid approach. In this study, a systematic and structured scenario development process of a hybrid approach was formulated. Based on this, sub-scenarios were extracted that describe the phenomena occurring in the repository over the performance period, categorized by period. By integrating and screening the extracted sub-scenarios, a scenario describing the phenomena occurring over the entire period of disposal was developed.

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Keywords: Spent nuclear fuel, Deep geological disposal, Scenario, Safety assessment, Hybrid approach

Development of a Robust Extensible Integrated Hydrologic Model for Safety Assessment of Geological Barrier Systems For the Final Disposal of Spent Nuclear Fuel

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The deep geologic repository (DGR) concept is widely accepted as the most feasible option for the final disposal of spent nuclear fuels. In this concept, a series of engineered and natural barrier systems are combined to safely store spent nuclear fuel and to isolate it from the biosphere for a practically indefinite period of time. Due to the extremely long lifetime of the DGR, the performance of the DGR relies especially on the natural geologic barriers. Assessing the safety of the DGR is thus required to evaluate the impacts of a wide range of geological, hydrogeological, and physicochemical processes including rare geological events as well as present water cycles and deep groundwater flow systems.

Due to the time scale and the complexity of the physicochemical processes and geologic media involved, the numerical models used for safety evaluation need to be comprehensive, robust, and efficient. This study describes the development of an accessible, transparent, and extensible integrated hydrologic models (IHM) which can be approved with confidence by the regulators as well as scientific community and thus suitable for current and future safety assessment of the DGR systems. The IHM under development can currently simulate overland flow, groundwater flow, near surface evapotranspiration in a modular manner. The IHM can also be considered as a framework as it can easily accommodate additional processes and requirements for the future as it is necessary. The IHM is capable of handling the atmospheric, land surface, and subsurface processes for simultaneously analyzing the regional groundwater driving force and deep subsurface flow, and repository scale safety features, providing an ultimate basis for seamless safety assessment in the DGR program. The applicability of the IHM to the DGR safety assessment is demonstrated using illustrative examples.

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Keywords: Deep geologic repository (DGR), Safety assessment, Integrated hydrologic model (IHM)

Estimation Method of Fracture Size Distribution Applicable to URL Site

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The distribution characteristics of rock fractures determine the hydro-mechanical behavior of natural barriers. Rock fractures are defined by various parameters, which are analyzed as the probability distribution from observation results by surveying the exposed rock surface or borehole. The size is known to have the most uncertainty among the fracture parameters because it cannot be directly measured. Therefore, various estimation methods have been proposed for fracture size distribution using the fracture traces observable on the rock surface. However, most methods are based on a planar survey area, limiting their applicability to the underground research laboratory (URL) excavated in the form of tunnels.

This study aims to review a method that can be applied to estimate the size distribution of fractures in deep rock masses at the URL site. The estimation method using the joint center volume (JCV) has recently been extended to be applicable regardless of the geometry of the survey area, which means that it can be applied to the URL site with complex structures. To apply the JCV-based estimation method to non-planar survey areas, JCV calculation using Monte Carlo simulation and estimation of fracture size distribution using the maximum likelihood method are required. In this study, we applied the JCV-based estimation method to a tunnel-shaped survey area to examine its applicability to the URL site. The error rates were analyzed when there were fracture sets with various orientations, size distributions, and maximum fracture sizes in the rock mass, and it was found to be less than 10% in all cases. This result indicates that the JCV-based estimation method can be used to estimate the fracture size distribution of the surrounding rock mass if accompanied by a reliable survey of fracture traces on the tunnel surface inside the URL site. Also, since there are no restrictions on the geometry of the survey area, we can continuously update the estimation results during the URL excavation process to increase reliability. The fracture size distribution is essential for constructing the discrete fracture network (DFN) model of the rock mass units at the URL site. In the future, the uncertainty for the fracture size in the DFN model is expected to be reduced by applying the JCV-based estimation method.

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Keywords: Natural barriers, Rock fractures, Size distribution, Joint center volume, Underground research laboratory

Low-frequency Acoustic Emission Characteristics From Meter-scale Fracturing of Rock-like Brittle Materials for Deep Geological Disposal System

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The acoustic emission (AE) method as a passive non-destructive monitoring technique is proposed for real-time monitoring of mechanical degradation in underground structures, such as deep geological disposal of high-level nuclear waste (HLW). This study investigates the low-frequency characteristics of AE signals emitted during the fracturing of meter-scale concrete specimens; uniaxial compression tests (UCT) in a lab scale and Goodman jack (GJ) tests in a 1.3 m-long concrete block were conducted while acquiring the AE signals using low-frequency AE sensors. The results indicate a sharp increase in AE energy emission at approximately 60% and 80% of the yield stresses in the UCT and GJ tests, respectively. The collected AE signals were primarily found in two frequency bands: the 4-28 kHz range and the 56-80 kHz range. High-frequency AE signals were captured more as the stress increased in the GJ tests, which was in contrast to the UCT tests. Furthermore, the AE signals obtained from the Goodman jack tests tended to lower RA values than the UCT results. This study presents unique experimental data with low-frequency AE sensors under different loading conditions, which provides insights into field-scale AE monitoring practices.

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Keywords: Acoustic emission, Low-frequency sensor, Meter-scale, Concrete, Goodman jack test

Optimum Conditions to Develop Sand-Gyeongju Bentonite Mixtures as Buffer Materials Satisfying THM Performance Criteria

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In order to reduce the area of the high-level radioactive waste (HLW) repository, a buffer material with high thermal conductivity is required. This is because if the thermal conductivity of the buffer material is high, the distance between the disposal tunnels and the deposition holes can be reduced. Sand, which is a natural material and has higher thermal conductivity than bentonite, is added to bentonite to develop an enhanced buffer material. For the sand-bentonite mixture, it is important which sand to use and how much to add because an enhanced buffer material should satisfy both hydraulic (H) and mechanical (M) performance criteria while improving thermal conductivity (T). In this study, we would like to show what type of sand and how much sand should be added to develop an enhanced buffer material by adding sand to Gyeongju bentonite, a representative bentonite in Korea. For this purpose, the thermal conductivity, hydraulic conductivity, and swelling pressure of the sand-Gyeongju bentonite mixture according to the sand addition rate were measured. It is more efficient to use silica sand with smaller particles than Jumunjin sand which is a representative sand in Korea as an additive for an enhanced buffer material than using the Jumunjin sand. In order for the sand-Gyeongju bentonite buffer material to satisfy both the hydraulic and mechanical performance criteria as a buffer material while increasing the thermal conductivity, it is judged that the optimum dry density is 1.7 g/cm^3 at least and the optimum sand addition rate is 10% at most.

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Keywords: High-level radioactive waste repository, Enhanced buffer material, Sand-bentonite mixture, Thermal conductivity, Hydraulic conductivity, Swelling pressure

~~Shear Strength of Compacted Bentonite Saturated With Saline Solutions Investigated Through Triaxial Compression Tests~~

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Withdrawal

A Numerical Investigation of the Disposal Efficiency Enhancement for a High-level Radioactive Waste Repository

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As Korea has relatively small land area and large population density compared to other countries considering the DGD concept such as Finland and Sweden, improvements of disposal efficiency in the viewpoint of the disposal area might be needed for the current disposal system to alleviate the difficulties of site selection for the HLW repository. In this research, we conduct a numerical investigation of the disposal efficiency enhancement for a high-level radioactive waste (HLW) repository through three design factors: decay heat optimization, increased thermal limit of buffer, and double-layer concept. In the optimized decay heat model, seven SNFs with the maximum and minimum decay heat depending on actual burn-up and cooling time are iteratively combined in a canister. Thermal limit of buffer is assumed as 100°C and 130°C for reference and high-efficiency repository concepts, respectively. By implementing an optimized decay heat model and a single-layer concept with a thermal limit of buffer set at 100°C, the disposal efficiency increases to 2.3 times of the improved Korean Reference disposal System (KRS⁺). Additionally, incorporating either an increased thermal limit of buffer to 130°C or a double-layer concept leads to a further 50% improvement in disposal efficiency. By integrating all three design factors, the disposal efficiency can be enhanced up to five times that of the KRS⁺ repository. Our analysis of rock mass stability reveals that increasing the thermal limit of buffer can generate rock spalling failure in a wider area. However, when accounting for the effect of confining stress by swelling of buffer and backfill using the Mohr-Coulomb failure criteria, the rock mass failure only occurred at the corner between the disposal tunnel and deposition hole when the thermal limit of buffer was increased and a single-layer concept was applied. The results given in this study can provide various options for designing the high-efficiency repository in accordance with the target disposal area and quality of the rock mass in the potential repository site.

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Keywords: High-level radioactive waste repository, Disposal efficiency enhancement, Decay heat optimization, Thermal limit of buffer, Double-layer concept

Mineralogical Characteristics of Fracture-filling Minerals and Aperture Distribution in a 1 m Scale Fractured Granite

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A disposal research program for HLW has been carried out since 1997 with the aim of establishing the preliminary concept of geological disposal in Korea. The preliminary studies were conducted by conducting manufacture and installation of an in-situ nuclide migration system in KAERI Underground Research Tunnel (KURT). Nuclides could be released from a deep underground disposal facility due to thermal and physicochemical changes into the surrounding environments. Understanding on the migration and retardation processes of nuclides in a fractured rock is very important in the safety assessment for the radioactive waste disposal. In this study, we evaluated fracture filling minerals and aperture distribution (3D map) along the fracture surfaces under the controlled conditions. The fractured granite block which has a single natural fracture of 1 m scale was sampled in a domestic quarry (Iksan), which groundwater had been flowed through. This rock has an interconnected porosity of 0.36 with the specific gravity of 2.57. The experimental set-up with the granite block with dimensions of 100×60×60 (cm). A flow of de-ionized water through the fracture between pairs of boreholes was initiated and the pressure required to maintain a steady flow was measured. In additions, fracture filling minerals were sampled and examined by mineralogical and chemical analyses. There are phyllosilicate minerals such as illite, kaolinite, and chlorite including calcite, which are fracture filling minerals. The illite and kaolinite usually coexist in the fracture, where their content ratio is different according to which mineral is predominant. For the evaluation of fracture, surface was divided into an imaginary matrix of 20×20 sub-squares as schematically. The calculated results are expressed as a two dimensional contour and a three dimensional surface plot for the aperture distribution in the fracture. The aperture value is distributed between 0.075 and 0.114 mm and the mean aperture value is 0.095 mm. The fracture volume is about 55 ml. Also the ^{137}Cs sorption (batch test) distribution coefficients increased to $K_d = 800\sim 860$ mL/g in the fractured rock because of the presence of secondary minerals formed by weathering processes, compared to the bedrock ($K_d = 750\sim 830$ mL/g). These results will be very useful for the evaluation of environmental factor affecting the nuclides migration and retardation.

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Keywords: Fracture-filling minerals, Aperture distribution, Fracture, Granite, Migration

3분과

고준위폐기물처분(Poster)

How to Design a Performance Assessment Methodology for a Double-layered Spent Nuclear Fuel Disposal System

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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 (SNFs) generated from each site are temporarily being stored as wet or dry storage type at each plant site. These SNFs with their high radiotoxicity, heat generating, and long-lived radioactivity are currently 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 management policies in consideration of their situations. In Korea long-term R&D research program for safe management of SNF has also been conducted during last couple of decades since around 1997, during which several various 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 SNF in stable crystalline rock at a depth of around 500 m which has been regarded as one of the most plausible method worldwide to direct disposal of SNF. The world first Finnish repository will be also this type. Since the characteristics of SNF 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 SNF) 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 SNF. 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 SNF 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 will be 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.

Keywords: Spent fuel, Alternative disposal, High efficiency repository, Matrix borehole, GoldSim, Safety assessment

Comparison of Engineered Barriers for Alternative Disposal Concepts for CANDU Spent Nuclear Fuels in Korea

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It is expected that around 576,000 bundles of CANDU spent nuclear fuels (SNF) will be generated from the four CANDU reactors located at the Wolsong site. The authors designed and proposed a reference disposal concept based on the KBS-3 type and KURT geological data in 2022. In addition, we have reviewed the literatures and selected four alternative disposal methods to develop the higher-efficiency disposal concept than the reference concept since 2021. As known well, the most important safety functions of the geological disposal are containment and isolation, and the secondary function is retardation. A disposal canister covers the former, and buffer may do the latter.

In this study, we design the engineered barrier systems for the four alternative concepts: (1) mined deep borehole matrix, (2) sub-seabed disposal, (3) deep borehole disposal, and (4) multi-level disposal. Assuming total 10,000 tU of CANDU SNF, four different kinds of unit disposal module consisting of disposal canisters and compacted bentonite buffers are designed based on the technique currently available. Two alternative concepts, sub-seabed disposal and multi-level disposal, share the same unit module design with the reference concept in 2022. For all the alternative concepts, we assume that the density of the compacted buffer is 1.6 g/cm^3 .

For the mined deep borehole matrix disposal, we introduce a disposal canister slightly modified from the Canadian NWMO canister with a capacity of 48 bundles. The thickness of a copper layer is changed to be 10 mm considering the long-term corrosion resistance. The buffer thickness around a disposal canister is 20 cm, and the diameter of a borehole is 100 cm. Two different kinds of buffer blocks are proposed for the easy handling of them. For the deep borehole disposal, a SiC-stainless steel canister is designed, and 63 bundles of CANDU SNF is emplaced in the canister. We expect that the SiC ceramic canister shows very excellent corrosion resistance and has a high thermal conductivity under the geological conditions. The deep borehole will be plugged with four layered sealing materials consisting of granite blocks, compacted bentonite, SiC ceramic, and concrete plugs.

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Keywords: CANDU spent nuclear fuel, Alternative disposal concepts, Engineered barrier, Disposal canister, Bentonite buffer

Strategies and Design Requirements for a High-efficiency Disposal Concept of PWR Spent Nuclear Fuels in Korea

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In order to use nuclear energy stably, high level radioactive waste including spent nuclear fuel that is inevitably discharged from nuclear power plants after electricity generation must be managed safely and isolated from the human living area for a long period of time. In consideration of the accumulated amount of spent nuclear fuel anticipated according to the national policy for HLW management, the area required for the deep geological repository facility is expected to be very large. Therefore, it is essential to conduct various studies to optimize the area required for the disposal of spent nuclear fuel in cases where the nationally available land is extremely limited, such as in Korea. In this study, as part of such research, the strategies and the requirements for the preliminary design of a high efficiency repository concept of spent nuclear fuel were established.

For PWR spent nuclear fuel, seven assemblies of spent nuclear fuel can be accommodated in a disposal canister, and high burnup of spent nuclear fuel was taken into consideration, and the source terms such as the amount and time of discharge and disposal were based on the 2nd national basic plan. By evaluating the characteristics, the amount of decay heat that can be accommodated in the disposal canister was optimized through the combination of seven assemblies of spent nuclear fuel. The cooling period of the radiation source for the safety assessment of the repository system was set at 55 years, and the operation of the repository would start from 2070 and then the disposal schedule would be conducted according to the disposal scenario based on the national basic plan.

With these disposal strategies described above, the main requirements for setting up the conceptual design of the high efficiency repository system to be carried out in this study were described below.

- A combination of seven spent nuclear fuels with high heat and spent nuclear fuels with low heat was loaded into a disposal canister, and the thermal limit per disposal canister was 1,600 W.
- In order to maintain the long-term performance of the repository, the maximum temperature design limit in the buffer material was set to 130°C.
- In the deep disposal environment, the safety factor [yield strength/maximum stress] required to maintain the structural stability of the disposal canister should be maintained at 2.0 or higher so that integrity of the canister can be maintained even under long-term hydrostatic pressure and buffer swelling pressure in the deep disposal environment.
- The repository should have a maximum exposure dose of 10 mSv/yr or less, which is the legal limit in case of a single event such as an earthquake, and the risk level considering natural phenomena and human intrusion, which is less than the legal limit of 10^{-6} /yr.

These strategies and requirements can be used to develop the high-efficiency geological disposal concept for spent nuclear fuels as an alternative disposal concept.

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Keywords: Spent nuclear fuels, High-efficiency, Alternative deep geological disposal system, Development strategy, Design requirements

Distribution Coefficients of Th(IV) Onto Granite and Biotite Gneiss in Gyeongju Groundwater

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Th(IV) is a stable actinide that can act as a chemical analogue of U(IV) and Pu(IV), which are important radionuclides in safety assessments of deep geological repositories (DGR). Therefore, to understand the geochemical behaviour of U(IV) and Pu(IV), batch sorption of Th(IV) onto crystalline rocks were performed in oxidising conditions. The distribution coefficients (K_d) of Th(IV) were of particular interest.

Gyeongju fresh groundwater (GF) and Gyeongju brackish groundwater (GB) were obtained at the Gyeongju Low and Intermediate Level Radioactive Waste (LILW) Disposal Facility. Crystalline granite (gr) and biotite gneiss (bg) were collected in Gyeongju and Gwacheon respectively and were grounded to a particle size smaller than 150 μm . Sorption samples were continuously shaken for 7 days under 200 rpm at 25°C. The liquid-to-solid ratio (V/m) was 200 $\text{L}\cdot\text{kg}^{-1}$. Th(IV) concentrations of the sorption samples were determined by UV-Vis-NIR absorption colorimetry from the formation of Th(IV)-arsenazo III complexes. Although the method allowed the initial Th(IV) concentrations to be determined, the final Th(IV) concentrations fell below the limit of detection (LOD), $6.27\times 10^{-9} \text{ mol}\cdot\text{L}^{-1}$. Taking the LOD as the final concentrations, conservative K_d were calculated to be 4,410 $\text{L}\cdot\text{kg}^{-1}$ for GF-gr and GF-bg, and 7,830 $\text{L}\cdot\text{kg}^{-1}$ for GB-gr and GB-bg.

The result indicates a strong sorption affinity of Th(IV) onto granite and biotite gneiss within Gyeongju groundwater, suggesting a similar behaviour for U(IV) and Pu(IV). Furthermore, comparison of the conservative K_d obtained from the experiment were compared with existing K_d values of Th(IV). Such analysis and comparison of Th(IV) K_d in various types of groundwater could help locate the optimal site for a DGR in South Korea.

Acknowledgements

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Keywords: Th(IV), Granite, Biotite gneiss, Batch sorption, Distribution coefficient, Arsenazo III

Literature Survey of Natural Environment and Human Society in Korea for Application to the HLW (SNF) Disposal Sites Biosphere Assessment

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IAEA safety standards document and international programs (such as BIOMASS) related to the assessment of the biosphere around High Level Radioactive Waste (including Spent Nuclear Fuel) repositories require the assessment of the biosphere to use the assumption that the current natural environment and human society will be maintained, and at the same time, the evolution of the distant future changes also need to be taken into account.

In Korea, which has not designated candidate disposal sites, it is necessary to investigate and predict the current state and future changes of the natural environment throughout Korea and apply it practically to Biosphere assessment (for BDCF derivation) for candidate disposal sites suitability assessment and Safety Case (for performance assessment) preparation for design, construction, operation, and post-closure management.

To this end, the natural environment in the fields of Topography, Geology, Soil, Ecology, Weather and Climate, Animals and Plants, Hydrology, Ocean, Land-use, etc. and human society in the fields of Population Distribution, Spatial-Planning, Urban Form, Industrial-Structure, Lifestyle etc. are being investigated in the context of current status, past change records, and future change potential in the Korean Peninsula. This paper summarizes those investigations to date.

This study referred Biomass-6 [IAEA] and National Atlas I (2019)/II (2020)/III (2021) [National Geographic Information Institute of the Korea Ministry of Land, Infrastructure and Transport].

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Keywords: SNF deep geological repository, Biosphere assessment, Korean peninsula, Natural environment, Human society

Comparison of Temperature Effects on Korean Bentonite Permeability

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Bentonite has been considered as a potential buffer material in the engineering barriers of high-level radioactive waste disposal systems. The intrusion of groundwater and heat from the waste change the temperature of the bentonite, which can alter the hydraulic properties of the bentonite. In this study, temperature effects on permeability were observed in two Ca-type bentonites. Laboratory tests were conducted on two types of block that were compacted using Korean Gyeongju bentonite and bentonil-WRK at different dry densities. Permeability tests were conducted at three different temperatures, namely 30°C, 60°C, and 90°C, using deionized water. Moreover, comparison between two Ca-type bentonites is carried out.

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Keywords: Bentonite, Hydraulic behavior, Permeability, Temperature

Comparing Biosphere Conceptual Models for Groundwater Release Scenarios in Deep Geological Disposal Safety Assessment

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Two sets of analyses for the cases of groundwater release to well and sea ecosystems were conducted for the environmental impact assessment of high-level radioactive waste disposal facilities. After obtaining the respective BDCF (Biosphere Dose Conversion Factor) results for the scenarios of well-farming and marine water fishing using different biosphere assessment conceptual models implemented in ECOLEGO, they were compared each other. The purposes of these analyses are to identify reference generic biosphere conceptual models and to get insight on model uncertainty.

In this study, the endpoint used for the comparison of the ECOLEGO biosphere models was the so-called Biosphere Dose Conversion Factor (BDCF), which is defined as the maximum value of the total dose to the exposed group, in Sv/yr, resulting from a continuous unit release of 1 Bq/yr during the whole simulation time either to the well compartment (BDCF_Well) or to the marine water compartment (BDCF_Sea). The radionuclides considered in the comparison were Cs-137, I-129, Nb-94, Ni-59, Ni-63, Sr-90 and Tc-99.

The conceptual models used in the biosphere assessment of the releases to a well are based on models that have been used by the DOE (simple-soil model) and SKB (complex-soil model) in safety assessments of radioactive waste repositories, respectively. Difference between two conceptual models used in the assessment of the releases to a sea is the number of compartments representing the sea; i.e., one model represents the sea with one compartment for the water and one for the sediment (single-compartment model), whereas the alternative model uses two compartments for the water and the sediments: one for the inner coast and one for the outer coast (double-compartment model).

The results of the BDCF_Well to a farmer obtained with the DOE and SKB models are shown to be very close to each other. Despite the differences in conceptual models and parameters, the results are within a maximum difference of a factor of 4. The results from the SKB model were higher for all radionuclides. The values of the BDCF_Sea obtained with the single- and double-compartment models are shown to be larger differences with a maximum order of 2. For all studied radionuclides, the double-compartment model produces higher BDCFs than does the single-compartment model. The differences would be due to activity concentrations in both water and sediments. Since the hydrodynamic behavior assumed for flow in the sea could significantly influence the dilution volumes and hence the concentrations, it is found that site-specific investigations are necessary to establish an appropriate marine biosphere conceptual model.

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Keywords: Biosphere assessment, Conceptual model, Well farming, Marine water fishing, Compartment modeling

Double Cold Spray Coating Technique for the Galvanic Corrosion Resistance of a Disposal Canister

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The purpose of this study was to examine whether galvanic corrosion of copper occurs by inserting a third barrier layer with a higher corrosion potential than copper between copper and cast iron when the copper layer is locally perforated by pitting or partial corrosion. A triple layer composed of copper, inserted metal, and carbon steel was manufactured by cold spray coating of inserting metal powders such as Ag, Ni, and Ti on carbon steel plate followed by Cu coating on it. First, the corrosion properties were evaluated electrochemically for each metal coating. As a result of Tafel plot analysis in KURT groundwater condition, the corrosion potential of Fe (-567 mV) was much lower than that of Cu (-91 mV), and the corrosion potential of Ni (-150 mV) was also lower than that of Cu. Therefore, Ni was likely to corrode before Cu. However, the corrosion current of Ni was lower than that of the Cu. In the galvanic specimen where the copper and inserting metal were exposed together, Cu-Fe was much lower corrosion potential of -446 mV, and the corrosion potential of Cu-Ti, Cu-Ni, and Cu-Ag were slightly higher than that of Cu. Therefore, it seemed that Ag, Ni, and Ti all might promote galvanic corrosion of surrounding copper when the copper layer was perforated to the inserted metal layer. If the metal insertion presented in this study operates properly, the disposal container does not need to worry about the partial corrosion or non-uniform corrosion of external copper layer.

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

Influence of Depth-dependent Rock Properties on Stress Field Determination in KURT Site

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In KAERI, a site descriptive model for stress field estimation had already been constructed by using integrated field data within KURT site scale. A sub-divided rock block domain containing major fracture zones has spatial rock mass and fault properties. The properties were decided based on the rock classification results of several borehole investigations. Modeling for maximum and minimum horizontal stress field estimation was performed and compared with the in-situ data. As a result, a depth-dependent stress ratio was adopted to obtain numerical results closer to actual in-situ data. Although the results were suitable at a relatively low depth (~500 m), there is still some deviation trend at a deep depth.

This study aims to improve these modeling results by incorporating not only depth-dependent stress ratio but also changes in rock mass properties along the depth. The deep borehole of DB2 in the KURT site indicated fracture distribution corresponding to the property changes. Natural fractures are typically randomly oriented, and the fracture frequency decreases with increasing depth. The increase in P-wave velocity log data accompanies these features. A discrete fracture network (DFN) model can be used to simulate fractured rock explicitly, but DFN modeling is not feasible for site scale analysis because of its numerical efficiency. Therefore, as a preliminary model in this study, the effect of fracture distribution was considered by substituting the influence for the depth-dependent property. The properties were estimated from the fracture frequency and P-wave velocity log data. The influence of elastic modulus and density on the site stress field was dominant, with decreasing the deviation trend between modeling and in-situ data at a deep depth. Considering that the depth of the repository construction is within about 500 m, it may not be necessary to consider the change of rock properties with depth. However, it was determined that the rock property effect might need to be considered when the loading conditions change due to subsidence in the long-term evolution scenario. Continuously, this site descriptive modeling will be interdependently conducted with a representative DFN block model for deriving equivalent properties in fractured rock.

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Keywords: Stress field estimation modeling, KURT site, Fracture frequency, Depth-dependent property

A Study on the Radiological Impact Assessment for a High-Level Radioactive Waste Disposal Site

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Currently, there are 25 nuclear power plants (NPPs) in operation in Korea, including 22 pressurized water reactors (PWRs) and three pressurized heavy water reactors (PHWRs). Two NPPs, including Kori Unit 1 and Wolsong Unit 1, are permanently shut down and awaiting decommissioning. If Kori Unit 2, which is expected to be permanently shut down soon, is included, the number of decommissioning NPPs will be increased to three. Spent fuels (SFs) are continuously generated during the NPP operation, which are stored in an SF storage pool in NPPs to cool down the decay heat emitted from SFs. For safe NPP operation, SFs must be regarded as waste, and a disposal site must be selected to isolate SFs. However, an appropriate site has yet to be selected in Korea.

SFs contain long-lived nuclides with a high specific activity. For disposal, it is important to characterize the nuclides in the fuels and delay the migration of the nuclides to the environment when SFs are placed in a future disposal facility. If the disposal container is broken, the nuclides in the fuels escape from the filling material, such as bentonite. These escaped nuclides are dissolved in groundwater and migrate to the surface of the earth. Thus, it is possible to assess the radiological impact, such as the exposure dose during and after the disposal, if the types and characteristics of nuclides in SFs are known.

This study investigated the nuclides in SFs and identified exposure scenarios that may occur in the disposal process of SFs and migration characteristics when the nuclides leak into groundwater to propose a dose assessment methodology for workers and the public.

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Keywords: Spent fuel, Disposal, Characterization, Dose assessment, Radiation protection

Analysis of Overseas Distribution Coefficients for Radionuclides Sorption on Buffer Material Under Aerobic Condition

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A disposal system for spent nuclear fuel divides into two parts; (1) engineered barriers including spent nuclear fuel, canister, buffer and backfill, (2) natural barriers surrounding engineered barriers. Sorption and diffusion are main retardation mechanisms for the migration of released radionuclides. We analyzed the sorption properties of radionuclides for bentonite as a buffer material and collected/evaluated the distribution coefficients for the purpose of safety assessment for the deep geological disposal of a spent nuclear fuel. Through this, we presented recommended distribution coefficients for radionuclides required for the safety assessment. This work included the radionuclides as follows; alkali and alkaline earth metals (Cs, Sr, Ba), lanthanides (Sm), actinides (Ac, Am, Cm, Np, Pa, Pu, Th U), transition elements (Nb, Ni, Pd, Tc, Zr), and others (C, Cl, I, Rn, Se, Sn). The sorption of radionuclides affected various geochemical conditions such as pH/carbonates, redox potential, ionic strength, radionuclide concentration, kinds and amounts of minerals, and microbes. Among the evaluated radionuclides, Cs, Ni, Pd, and Ra is sensitive to the ionic strength, while Np, Pu, U, Se, and, Tc is sensitive to the redox condition. 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 were suggested.

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

A Quantitative Analysis of the Applicability of Richards' Flow Approach for High-Temperature Repository Concept

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Since 1992, various numerical codes, such as TOUGH-FLAC and ROCMAS, have been developed and validated to dispose of Spent Nuclear Fuel (SNF) safely through a series of DEvelopment of COupled models and their VALidation against EXperiments (DECOVALEX) projects. These codes have been developed using different approaches, such as general two-phase flow and Richards' flow which is an approximated approach neglecting gas pressure change, to implement the same multi-physics behaviors. However, the quantitative analysis for numerical results, which originated from different fundamental approaches, has not been conducted accurately. As a result, improper utilization of the approach to analyze certain conditions occurring such as dramatic gas pressure change may result in erroneous outcomes and systemic problem pertaining to TH analysis. In this study, the quantitative analysis of the two approaches, in terms of TH behavior, was conducted by comparing them with a 1D simulation of the CTF1 experiment carried out by laboratory experiment. The results calculated by different approaches show agreement in terms of TH behaviors and material properties change until 120°C. The results verify the applicability of Richards' flow approach in a high temperature environment above the current thermal criteria, set as 100°C, and gas pressure change does not have a significant impact until 120°C. Therefore, although further studies for applicability of Richards' flow are needed to suggest the appropriate temperature range, these quantitative analyses may contribute to the performance assessment of a compact repository using the high-temperature bentonite concept, which is currently gaining attention.

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Keywords: General two-phase approach, Richards' flow approach, Quantitative analysis, CTF1 experiment

Simulation of Geochemical Evolution of Bentonite Buffer Using an Adaptive Process-based Total System Performance Assessment Framework (APro)

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In the engineered barrier system of deep geological disposal repository, complex physicochemical phenomena occur throughout the entire disposal time, consequently impacting the safety function. The bentonite buffer, a significant component of the engineered barrier system, can be geochemically altered due to the changes in host rock groundwater, temperature, and redox condition. Such changes may have direct or indirect effects on radionuclide migration in case of canister failure. Therefore, a modeling tool that accounts for coupled thermal-hydraulic-mechanical-chemical (THMC) processes is necessary for the safety assessment.

To this end, the Korea Atomic Energy Research Institute (KAERI) has developed the APro, a modeling interface for conducting safety assessment of deep geological disposal repository. The APro considers coupled THMC processes that influence radionuclide migration. Here, the solute transport considering thermal and hydraulic processes are calculated using the COMSOL multi-physics, while geochemical reactions are carried out in PHREEQC. The two software are coupled using a sequential non-iterative operator splitting approach, and transport of non-water H, non-water O, and charge were additionally considered to enhance the coupling model stability. Finally, the applicability of APro to simulate long-term geochemical evolution of bentonite was demonstrated through benchmark studies to evaluate the effects of mineral precipitation/dissolution, temperature, redox, and seawater intrusion.

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Keywords: Safety assessment, APro, COMSOL, PHREEQC, Geochemical evolution

Simulation Analysis of Petrogenesis in KURT Area for Development of Long-term Evolution Reconstruction Methodology of Natural Barrier

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A methodology is under development to reconstruct and predict the long-term evolution of the natural barrier comprising the site of radioactive waste disposal. The natural barrier must protect the human zone from radionuclides for a long time. So for this, we need to be able to restore the evolution of the bedrock constituting the natural barrier from the past to the present and to predict from the present to the future. A methodology is being studied using surface outcrop, tunnel face of KURT (KAERI Underground Research Tunnel), and drill core at KAERI (Korea Atomic Energy Research Institute). Among them, drill core is an essential material for identifying deep geological properties, which could not be confirmed near the surface when considering the geological condition of the repository in the deep part. In this study, we selected several qualitative and quantitative analyses to construct a deep lithological model from the disposal perspective. These were applied to drill core samples around the KURT. There are the dikes presumed the Cretaceous were intruded by Jurassic granitoids in the study area. Analyzing trace elements of each rock type in the study area classified through geochemical characteristics and microstructure in previous studies made it possible to obtain qualitative information on the petrogenetic process. In addition, synthesizing the quantitative numerical age allows for grasping the evolution of bedrock, including intrusion and cutting relationships. LA-ICPMS was used for determining the age of zircons in plutonic rocks. The highly reliable ^{40}Ar - ^{39}Ar method was selected for volcanic rocks because it can correct the loss of Ar gas and obtain the values of two types of Ar isotopes in a single sample. As a result, it was possible to infer the formation environment of rocks through anomalies in specific trace element content. And according to the numerical ages, it was possible to support the known separated rock type found in previous studies or to present a quantitative precedence relation for unclassified rocks. These methods could be applied to reconstruct the long-term evolution of bedrock within natural barriers.

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Keywords: Petrogenesis, Age dating, Trace element, Natural barrier, Long-term evolution

Relationship Among the Surface Environment Components Influenced by Uplift-Subsidence and Erosion-Deposition

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Performance and safety assessments for deep geological disposal are often conducted over a long-term time scale, such as from hundreds of thousands to a million years. During this period, it is expected that the surface environment will be changed significantly. Uplift-subsidence and erosion-deposition are thought to be included as the main causes of the changes, and it is necessary to evaluate their expected effects.

In this study, the conceptual processes of the changes in the surface environment components were to be presented by identifying the uplift-subsidence and erosion-deposition processes and analyzing their effect on the surface environment components.

For inferring the long-term change process of the surface environment due to the internal activities of the Earth, the process of uplift and subsidence caused by crustal movements that change the subsurface environment through the deep and shallow underground was briefly presented in the form of a chain flowchart. Uplift-subsidence is mainly caused by diastrophism due to tectonic movement, such as subduction at the boundary of plates. They can change the geomorphology by affecting sea-level change and erosion-deposition. The changed geographical features have an influence on the distribution of surface water and the flow path of groundwater. They also have an impact on the scale and processes of local uplift and erosion, which can be the main factors of pedogenesis and vegetation in the local site.

The results of this study can be helpful for formulating scenarios related to long-term evolution in the surface environment required for performance and safety assessments of deep geological disposal.

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Keywords: Long-term evolution, Surface environment, Uplift-subsidence, Erosion-deposition, Geological disposal, Radioactive wastes

Study on the Swelling Pressure Characteristics of Bentonil WRK-silica Sand Mixed Backfill Material

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Backfill is one of the key elements of deep geological disposal. The backfill material is used to fill disposal tunnels and is mainly composed of swellable clay, preventing the migration of nuclide and structurally supporting the tunnel. The selection and application of backfill material are critical for the stable and efficient disposal of spent fuel. Therefore, it is essential to secure various candidate materials for backfill and to comprehensively understand the properties and behavior of these materials. Recently, the Korea Atomic Energy Research Institute has selected a candidate material called Bentonil-WRK and is evaluating its applicability. To utilize this material as backfill, the safety function of a mixed backfill concept, consisting of sand and Bentonil-WRK, was assessed. The swelling pressure was measured as a function of dry density for a bentonite/silica sand mix ratio of 3/7. The results showed that the swelling pressure ranged from 0.15 to 0.273 MPa, depending on the dry density, with higher dry densities resulting in higher swelling pressures. The measured swelling pressure met the target performance criteria suggested by SKB and Posiva (i. e., 0.1 MPa), but did not meet the design requirement for swelling pressure (i. e., 1 MPa). This indicates the need for further research after increasing the mass fraction of bentonite (e. g., mix ratio 4/6 or more). The results of this study are expected to be used in the selection of candidate backfill materials and the establishment of design guidelines for engineered barrier backfill.

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Keywords: Disposal repository, Backfill, Swelling pressure, Hydraulic conductivity

Sensitivity Analysis for Two-phase Flow System in the FE Experiment Using OGS-FLAC

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Geologic disposal at deep depth is an acceptable way to dispose of high-level radioactive waste and isolate it from the biosphere. The geological repository system comprises an engineered barrier system (EBS) and the host rock. The system aims to delay radionuclide migration through groundwater flow, and also, the flow affects the saturation of the bentonite in the EBS. The thermal conductivity of bentonite is a function of saturation, so the temperature in the EBS is directly related to the flow system. High-temperature results in the two-phase flow, and the two-phase flow system also affects the flow system. Therefore, comprehending the influencing parameters on the flow system is critical to ensure the safety of the disposal system.

Various studies have been performed to figure out the complex two-phase flow characteristics, and numerical simulation is considered an effective way to predict the coupled behavior. DECOVALEX (DEvelopment of COupled models and their VALidation against EXperiments) is one of the most famous international cooperating projects to develop numerical methods for thermo-hydro-mechanical-chemical interaction, and Task C in the DECOVALEX-2023 has the purpose of simulating the Full-scale Emplacement (FE) experiment at the Mont-Terri underground research laboratory. We used OGS-FLAC, a self-developed numerical simulator combining OpenGeoSys and FLAC3D, for the simulation and targeted to analyze the effecting parameters on the two-phase flow system.

We focused on the parameters of bentonite, a key component of the disposal system, and analyzed the effect of compressibility and air entry pressure on the flow system. Compressibility is a parameter included in the storage term, defining the fluid storage capacity of the medium. While air entry pressure is a crucial value of the water retention curve, defining the relation between saturation and capillary pressure.

From a series of sensitivity analyses, low compressibility resulted in faster flow due to low storage term, while low air entry pressure slowed flow inflow into the bentonite. Low air entry pressure means the air easily enters the medium; hence the flow rate becomes lower based on the relative permeability definition. Based on the sensitivity analysis, we further investigate the effect of shotcrete around the tunnel and excavation damaged zone. Also, long-term analysis considering heat decay of the radioactive waste will be considered in future studies.

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Keywords: Two-phase flow, FE experiment, DECOVALEX, OGS-FLAC

Microstructure and Corrosion Resistance of Applicable Welding Methods for Copper Canister of Deep Geological Disposal

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Since spent nuclear fuel (SNF) should be isolated from the human life zone for at least 10^6 years, deep geological disposal (DGD) is considered a strong candidate for SNF management in many countries. Therefore, a disposal canister should be nearly immune to corrosion in such a long-term storage environment. Even though copper has a low corrosion rate of a few millimeters per million years in geological environments, the corrosion resistance of the copper welds must be preferentially validated, which inevitably occurs during the sealing of the disposal canister after the SNF is loaded. This is because the weld zone is a discontinuous area of microstructure, which can accelerate uniform and localized corrosion. In this study, the microstructural characteristics of copper welds in different welding conditions such as friction stir welding, electron beam welding, cold spray, were analyzed, focusing on the formation of microstructure, which affects resistance to corrosion. In addition, the microstructure and corrosion properties of the copper weld zone manufactured by recent wire-based additive manufacturing (AM) technology were experimentally evaluated. From this preliminary test result, it was found that the corrosion characteristics of the welds produced by the AM process using wire are comparable to those of the conventional forged copper plate.

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Keywords: Disposal canister, Copper welds, Deep geological disposal, Corrosion, Additive manufacturing

Montmorillonite Alteration in Alkaline and Saline Solutions and Its Retention Capacity for Cesium and Iodide

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Bentonite is a potential buffer material of multi-barrier systems in high-level radioactive wastes repository. Montmorillonite, the main constituent of the bentonite, is 2:1 type aluminosilicate clay mineral with high swelling capacity and low permeability. Montmorillonite alteration under alkaline and saline conditions may affect the physico-chemical properties of the bentonite buffer. In this study, montmorillonite alteration by interaction with synthetic alkaline and saline solution and its retention capacity for cesium and iodide were investigated. The experiments were performed in three different batches (Milli-Q water, alkaline water, and saline water) doped with cesium and iodide for 7 days. Alteration characteristics and nuclide retention capacity of original- and reacted bentonite was analyzed by X-ray diffraction (XRD), Fourier Transform Infrared (FTIR) spectroscopy, Scanning Electron Microscope (SEM), Nuclear Magnetic Resonance (NMR) and Cation Exchange Capacity (CEC) analysis. From the results, cesium retention occurred differently depending on the presence of competing ions such as K, Na, and Mg ions in synthetic solutions, while iodide was negligibly removed by bentonite. Montmorillonite alteration mainly occurred as cation exchange and zeolite minerals such as merlinoite and mordenite were new-formed during alkaline alteration of the montmorillonite. CEC value of reacted bentonite increased by formation of the zeolite minerals under alkaline conditions.

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Keywords: Montmorillonite, Alteration, Alkaline, Saline, Cesium, Retention

Thermodynamic Modeling of Selenite Sorption Onto Ca-type Bentonil WRK Montmorillonite

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The safe disposal of high-level radioactive waste (HLW), including the discharged spent nuclear fuel (SNF) and contaminated by-products produced from relevant chemical treatments, has become a serious pending problem for numerous countries that operate the nuclear power plants. The deep geological disposal (DGD) has thus far been considered the most proven and viable solution for isolation of the HLW and preventing any significant release of radionuclides into the biosphere. The DGD system consists of the multiple engineered and natural barrier components. Among them, the montmorillonite-based buffer and tunnel backfills are designed to perform the two major geochemical functions: 1) preventing the ingress of groundwater and any chemicals that compromise the safety of waste canister and 2) retarding the migration of released radionuclides by providing sufficient chemisorption sites.

Therefore, it is essential to investigate the sorption mechanism of radionuclides onto montmorillonite and develop a thermodynamic reaction model in advance in order to accurately predict the long-term performance of engineered barriers and to reduce the uncertainties in the safety assessment of a deep geological repository (DGR) ultimately; thus far, sorption of chemical species onto mineral adsorbents has been widely described based on the concept of sorption-desorption distribution coefficient (K_d), the value of which is intrinsically conditional, and active scientific efforts have been made to develop robust thermodynamic sorption models which offer the potential to improve confidence in demonstration of radionuclide migration under a wide range of geochemical conditions.

The natural montmorillonites are generally classified into Na-type or Ca-type according to its exchangeable cation, and the Ca-montmorillonite containing clays are being considered as candidate materials for the engineered barriers of DGR in several countries; they generally have advantages of higher thermal conductivity and lower price than the Na-montmorillonite based clays, but their sorption capacities are still comparable. In this framework, we aimed to investigate the chemical interactions of Ca-montmorillonite with selenite [Se(IV)], which is a major oxyanionic species in terms of HLW disposal, and develop a reliable thermodynamic sorption model (TSM). The present work summarizes the characterization of Ca-montmorillonite separated from the newly adopted reference bentonite (Bentonil-WRK) by means of XRD, BET, FTIR, CEC measurement, and acid-base titration. Further, its sorption behaviors with aqueous selenite species under aqueous conditions of $S/L = 5$ g/L, $I = 0.01\text{--}0.1$ m CaCl_2 , $\text{pH} = 4.5\text{--}8.5$, $\text{pCO}_2 = 10^{-3.5}$ atm, and $T = 25^\circ\text{C}$ were examined, and the resulting thermodynamic data are discussed as well.

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Keywords: Radionuclide migration, Engineered barrier, Montmorillonite, Selenite, Sorption, Thermodynamic Modeling

Suggesting Simplified Reactions for Pyrite and Copper(I) Oxidation to Predict Corrosion Environments in Deep Geological Repository

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Chemical environments of near-field (Engineered barrier and surrounded host rock) can influence performance of a deep geological repository. The chemical environments of near-field change as time evolves eventually reaching a steady state. During the construction of a deep geological repository, O_2 will be introduced to the deep geological repository. The O_2 can cause corrosion of Cu canisters, and it is important predicting remaining O_2 concentration in the near-field. The remaining O_2 concentration in the near field can be governed by the following two reactions: oxidation of Cu(I) from oxidation of Cu and oxidation of pyrite in bentonite and backfill materials. These oxidation reactions (Cu(I) and pyrite oxidation) can influence the performance of the deep geological repository in two ways; the first way is consuming oxidizing agents (O_2) and the second way is the changing pH in the near-field and ultimately influencing on the mass transport rate of radionuclides from spent nuclear fuel (failure of canisters) to out of the engineered barrier. Hence, it is very important to know the evolution of chemical environments of near-field by the oxidation of pyrite and Cu. However, the oxidation kinetics of pyrite and Cu are different in the order of $1E7$ which means the overall kinetics cannot be fully considered in the deep geological repository. Therefore, it is important to develop a simplified Cu and pyrite oxidation kinetics model based on deep geological repository conditions. Herein, eight oxidation reactions for the chemical species Cu(I) were considered to extract a simplified kinetic equation. Also, a simplified kinetics equation was used for pyrite oxidation. For future analysis, simplified chemical reactions should be combined with a Multiphysics Cu corrosion model to predict the overall lifetime of Cu canisters.

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Keywords: Pyrite oxidation, Cu(I) oxidation, Deep geological repository, Copper canister, Finite element method

Conceptual Model Development for a Biosphere Dose Assessment Module in a Performance Assessment Tool (the APro)

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The Korean Nuclear Safety and Security Commission has established a general guideline for the disposal of high-level waste, which requires that radiological effects from a disposal facility should not exceed the regulatory safety indicator, a radiological risk. The post-closure safety assessment of the disposal facility aims to evaluate the radiological dose against a representative person, taking into account nuclide transport and exposure pathways and their corresponding probabilities.

The biosphere is a critical component of radiation protection in a disposal system, and the biosphere model is concerned with nuclide transport through the surface medium and the doses to human beings due to the contaminated surface environment. In past studies by the Korea Atomic Energy Research Institute (KAERI), the biosphere model was constructed using a representative illustration of surface topographies and groundwater conditions, assuming that the representative surface environment would not change in the future. Each topography was conceptualized as a single compartment, and distributed surface contamination over the geometrical domain was abstracted into 0D. As a result, the existing biosphere model had limitations, such as a lack of quantitative descriptions of various transport and exposure pathways, and an inability to consider the evolution of the surface environment over time. These limitations hinder the accurate evaluation of radiological dose in the safety assessment.

To overcome these limitations, recent developments in biosphere modeling have incorporated the nuclide transport process over a 2D or 3D domain, integrating the time-dependent evolution of the surface environment. In this study, we reviewed the methodology for biosphere modeling to assess the radiological dose given by distributed surface contamination over a 2D domain. Based on this review, we discussed the model requirements for a numerical module for biosphere dose assessment that will be implemented in the APro platform, a performance assessment tool being developed by the KAERI. Finally, we proposed a conceptual model for the numerical module of dose assessment.

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

A Phenomenological Review on the Dissolution of Montmorillonite by Strong Alkaline Cement Leachates and Suggestion on Requisites for the Assessment in Korea

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In the deep geological repository, a considerable quantity of cementitious materials is generally used for structural stability of subcomponents such as grout and concrete plug of disposition tunnel. Strong alkaline leachates (pH>13) are produced after cement is dissolved by groundwater inflow from bedrock. When the leachates are transported to bentonite porewater (e.g. buffer and backfill) and thereby water exchange occurs, the physical properties of bentonite such as swelling capacity and hydraulic conductivity are changed, which eventually affects the safety function and long-term stability of engineered barrier system (EBS). Thus, in this paper, we reviewed the performance assessment methodology for cement-bentonite interaction in the operating license application for the Finnish deep geological repository, and suggested what to prepare for the analysis on the domestic disposal facility.

In Finland, thermal-hydraulic-chemical analysis for dissolution of montmorillonite by alkaline leachates resulting from cement degradation during the saturation of bentonite was carried out using PRECIP code. From this analysis, it was confirmed that effect on pH was considered to be more significant than that on temperature and bentonite saturation. As a result of this analysis, it was predicted that all primary minerals (including montmorillonite, quartz, and calcite) were dissolved and some secondary minerals (e.g. chalcedony and celadonite) was precipitated by alkaline cement leachates transported to the bentonite. In addition, it was shown that silica was preferentially released while the montmorillonite was dissolved, thus cementation of the bentonite was occurred. Through this phenomenon, the swelling capacity of bentonite is reduced and the hydraulic conductivity of bentonite is increased, which have a significant impact on the performance of the buffer and backfill.

Considering this, study on spreading of alkaline leachates, which is a condition for dissolution of montmorillonite, is necessary for the performance assessment of the domestic deep geological repository. However, this requires the site-specific data for the following in the disposal site: (a) distribution in fractured bedrock and pore structure (e.g. porosity, pore size distribution and pore morphology) in the bedrock, (b) hydraulic gradient and salinity concentration of groundwater, and (c) flux and velocity of groundwater.

Results of this study is considered to be directly utilized to the conceptual design and performance assessment of the deep geological repository in Korea, provided that additional data on microbiological properties of groundwater are obtained for the site selected.

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Keywords: Cement leachate, Montmorillonite dissolution, Cement-bentonite interaction, Buffer and backfill, Deep geological repository

Suggestion on What to Prepare for the Performance Assessment for the Deep Geological Facility in Korea: Focusing on Buffer Mass Loss

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In buffer, a main component of engineering barrier system (EBS) in the deep geological repository, mass loss is mainly caused by upheave and mechanical erosion. The former is a phenomenon that bentonite in the upper part of the buffer moves to the backfill region due to groundwater intake and swelling. And, the latter is a phenomenon that bentonite on the surface of the buffer moves to the backfill region due to groundwater flow at the interface with host rock as the buffer saturates. Buffer mass loss adversely affects the fulfilment of the safety function of the buffer that is to limit and retard radionuclide release in the event of canister failure. Accordingly, in this paper, we reviewed how to consider this phenomenon in the performance assessment for the operating license application in Finland, and tentatively summarized data required to conduct the analysis for the domestic facility based on the review results.

Regarding buffer mass loss, the previous studies carried out in Finland are categorized as follows: 1) experiment on the amount of buffer upheave with groundwater inflow rate (before backfilling), 2) analysis for the amount of buffer upheave with groundwater inflow rate (after backfilling), 3) analysis of buffer erosion rate with groundwater inflow rate, 4) analysis for distribution of the groundwater inflow rate into the buffer for all deposition holes (using ConnectFlow modeling results), and 5) analysis of buffer mass loss with groundwater salinity. Finally, the buffer mass loss distribution table was derived from the results of 1) through 3) by combining with that of 4).

Given these studies, the following will be required for the performance assessment for buffer mass loss in the domestic disposal facility: a) distribution table of buffer mass loss for combined interactions taking into account effect of 5) (i.e. 1), 2), 3), and 5) + 4)), and b) Threshold for buffer mass loss starting to negatively affect the fulfilment of the safety function of the buffer.

Even though it is judged that the results of this study could be directly applied to developing the design concept of EBS and to conducting the performance assessment in the domestic disposal facility, it is essential to prepare a set of input data reflecting the site-specific design features (e.g. dimension, material used, site, etc.), which include saturation time and groundwater salinity.

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Keywords: Buffer mass loss, Buffer upheave, Mechanical erosion, Performance assessment

Phenomenological Analysis for Water Exchange in Bentonite Porewater and Suggestion on the Performance Assessment on the Domestic Disposal Facility for High-level Radioactive Waste

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Bentonite, a material mainly used in buffer and backfill of the engineering barrier system (EBS) that makes up the deep geological repository, is a porous material, thus porewater could be contained in it. The porewater components will be changed through ‘water exchange’ with groundwater as time passes after emplacement of subsystems containing bentonite in the repository. ‘Water exchange’ is a phenomenon in which porewater and groundwater components are exchanged in the process of groundwater inflow into bentonite, which affects swelling property and radionuclide sorption of bentonite. Therefore, it is necessary to assess conformity with the performance target and safety function for bentonite. Accordingly, we reviewed how to handle the ‘water exchange’ phenomenon in the performance assessment conducted as part of the operating license application for the deep geological repository in Finland, and suggested studies and/or data required for the performance assessment of the domestic disposal facility on the basis of the results.

In the previous assessment in Finland, after dividing the disposal site into a number of areas, reference and bounding groundwaters were defined considering various parameters by depth and climate change (i.e. phase). Subsequently, after defining reference and bounding porewaters in consideration of water exchange with porewater for each groundwater type, the swelling and radionuclides sorption of bentonite were assessed through analyzing components of the reference porewater. From the Finnish case, it is confirmed that the following are important from the perspective of water exchange: (a) definition of reference porewater, and (b) variations in cation concentration and cation exchange capacity (CEC) in porewater.

For applying items above to the domestic disposal facility, the site-specific parameters should be reflected for the following: structure of the bedrock, groundwater composition, and initial components of bentonite selected. In addition, studies on the following should be required for identifying properties of the domestic disposal site: (1) variations in groundwater composition by subsurface depth, (2) variations in groundwater properties by time frame, and (3) investigation on the bedrock structure, and (4) survey on initial composition of porewater in selected bentonite

The results of this study are presumed to be directly applied to the design and performance assessment for buffer and backfill materials, which are important components that make up the domestic disposal facility, given the site-specific data.

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Keywords: Performance assessment, Water exchange, Deep geological repository, Porewater

A Study on Enhancing Keyword Extraction Productivity Using Natural Language Processing for Metadata Generation of FEPs

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The development of Features, Events, and Processes (FEPs) and scenarios, which consider the long-term evolution of repository, is underway, along with the construction of input data and a model database for the adaptive process-based total system performance assessment framework, APro. PAPIRUS serves as an integrated information processing platform, enabling users to seamlessly access, search, and extract essential information. To enhance data usability, it is crucial to establish well-structured metadata for each dataset.

Regarding FEPs, individual FEPs consist of extensive text-based data and sets of other short textual data. To enhance the searchability of these FEPs, precise keywords must be assigned to each FEP. For user convenience, the PAPIRUS FEP database contains several FEPs not only the long-term evolution FEPs developed by KAERI but also thousands of FEPs from the databases such as NEA PFEPs and Posiva FEPs. Generating keywords for thousands of FEPs proves to be a labor-intensive task. Consequently, this study explores natural language processing techniques for keyword analysis to boost the productivity of the keyword generation process. Specifically, we employ Generative Pre-trained Transformer (GPT) models for keyword extraction.

Our test results for keyword extraction demonstrate that, although not flawless, providing suitable prompts yields sufficiently useful keyword sets. We identified several optimal prompts and developed an Excel-based program to derive keywords from the existing FEP database using these prompts. By using the outcomes of this study, initial versions of keyword sets for thousands of FEPs can be rapidly produced and subsequently refined through expert review and editing. The generated keywords will serve as metadata within PAPIRUS.

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Keywords: Features, Events, and processes (FEPs), Keyword extraction, Natural language processing, PAPIRUS

A Review on Considerations for the Performance Assessment Related to Microbial Activity in the Domestic Disposal Facility Based on Finnish Case Study

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A variety of microorganisms are contained in the groundwater and surrounding environment at the depth of a deep geological repository, and could adversely affect the integrity and/or safety of the facility under certain thermal, hydraulic and chemical conditions. In particular, microbial activity (in the buffer and backfill) around the canister can cause corrosion of the canister through sulfide production by sulfate-reducing bacteria (SRB), and subsequently promote radionuclide release through the corroded part. Namely, this phenomenon is important in a perspective of performance assessment since it will have an impact on the post-closure exposure dose in the biosphere by accelerating radionuclide leakage into the near-field due to deterioration of the canister integrity

In Finland, the performance assessment on microbial activity in buffer, backfill, and plug was performed for the licensing. However, in Korea, researches relevant to microbial activity are only in the early stage as of now. Accordingly, in this study, we draw initial considerations for the performance assessment on the phenomenon in the domestic facility based on review results for the methodology carried out as part of operating license application (i.e. SC-OLA). Studies on the performance assessment of microbial activity in Finland were mainly performed: (a) to investigate complex interactions among microorganisms in the repository by analyzing both indigenous and exogenous microorganisms through drilling, geological and geochemical analysis, (b) to identify microbial interactions at the buffer, backfill, and host rock interface for specific microorganisms that may affect activity of other microorganisms and integrity of the repository, (c) to analyze canister corrosion caused by microbial activity, particularly sulfide production by SRB, and (d) to characterize microbial illitization of montmorillonite that could affect permeability, hydraulic conductivity, and structural integrity of the repository.

From reviewing studies above, it is judged that studies labelled as (b) through (d) are applicable to the performance assessment of microbial activity for the domestic facility regardless of specific conditions in Korea. However, for study labelled as (a), the following data on reflecting domestic conditions should be additionally obtained: (1) radionuclide inventory and temperature in spent nuclear fuel, (2) swelling pressure and organic carbon content of bentonite, and (3) size, shape, and gas composition of pores in bentonite.

Results of this study could be directly applied to the design and performance assessment for buffer and backfill components, provided that input data specific to the domestic disposal facility is prepared for the assessment required.

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Keywords: Microbial activity, Microorganism, Performance assessment, Deep geological disposal

Numerical Modeling of Nuclide Transport Utilizing Reactive Tracers Within Fractured Rock

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With the increasing demand for a repository to safely dispose of high-level radioactive waste (HLW), it is imperative to conduct a safety assessment for HLW disposal facilities for ensuring the permanent isolation of radionuclides. For this purpose, the Korea Atomic Energy Research Institute (KAERI) is currently developing the Adaptive Process-based total system performance assessment framework for a geological disposal system (APro). A far-field module, which specifically focuses on fluid flow and radionuclide transport in the host rock, is one of several modules comprising APro. In Korea, crystalline rock is considered the host rock for deep geological disposal facilities due to its high thermal conductivity and extremely low permeability. However, the presence of complex fracture system in crystalline rock poses a significant challenge for managing fluid flow and nuclide transport.

To address this challenge, KAERI is participating in DECOVALEX-2023 Task F1, which seeks to compare and verify modeling results using various levels of performance assessment models developed by each country for reference disposal systems. Through the benchmark problems suggested by DECOVALEX-2023 Task F1, KAERI adopts the Discrete Fracture-Matrix (DFM) as the primary fracture modeling approach.

In this study, the transport processes of reactive tracers in fractured rock, modeled with DFM, are simulated. Specifically, three different tracers (conservative, decaying, adsorbing) are introduced through the fracture under identical injecting conditions. Thereafter, the breakthrough curves of each tracer are compared to observe the impact of reactive tracers on nuclide transport. The results of this study will contribute to a better understanding of nuclide behavior in subsurface fractured rock under various conditions.

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Keywords: APro, Fracture, DECOVALEX-2023, Nuclide transport, Decay, Adsorption

Life Prediction of Monitoring Piezoelectric Sensor Under Temperature Condition of HLW Repository

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The high-level nuclear waste (HLW) repository is a 500-1,000 m deep underground structure to dispose high-level nuclear waste. The waste has a very long half-time and is exposed to a number of stresses, including high temperatures, high humidity, high pressure. These stresses cause the structure to deteriorate and create cracks. Therefore, structural health monitoring with monitoring sensors is required for safety.

However, sensors could also fail due to the stresses, especially high temperature. Given that the sensors are installed in the bentonite buffer and the backfill tunnel, it is impossible to replace them if they fail. That's why it is necessary to assess the sensors' durability under the repository's environmental conditions before installing them.

Accelerated life test (ALT) can be used to assess durability or life of the sensors, and it is important to obtain the same failure mode for reliability tests including ALT. Before conducting the test, the proper stress level must be designed first to get reliable data in a short time. After that, acceleration of life reduction with increasing temperature and temperature-life model should be determined with some statistical methods. In this study, a methodology for designing stress levels and predicting the life of the sensor were described.

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Keywords: High-level nuclear waste repository, Monitoring sensor, Accelerated life test, Failure mode, Life prediction

Model Concept for Simulating Mechanical Behavior of Deep Geological Repository

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For the performance analysis of deep geological repository systems, numerical simulation with multi-physics is required, which specifically covers Thermal (T), Hydraulic (H), and Mechanical (M) behaviors in the disposal environment. Numerous simulation models have been developed so far, each of which varies in the approach and methodology for solving THM problems. Fully-coupled THM simulation codes such as ROCMAS, THAMES, and CODE_BRIGHT were mainly developed in the initial stage of DEvelopment of COupled models and their VALidation against EXperiments (DECOVALEX), with the advantage of thorough calculations consisting of correlated several variables on different physics. Due to the difficulty of solving the complex Jacobian Matrix and the following burden for the computational calculation, weakly-coupled THM models have been suggested in recent researches: TOUGH2-MP with FLAC3D, TOUGH2 with UDEC and OpenGeoSys with FLAC3D. This methodology of loose coupling allows the practical use of computational code optimized for each physics, thereby increasing the efficiency in simulation. However, these suggested models require two different numerical codes to calculate THM behaviors, which leads to several inherent issues: compatibility during maintenance, updating and dependency between two codes. In this study, therefore, the authors build a unified code for simulating THM behaviors in the deep geological repository. The concept involves the iterative sequential coupling between TH and M for calculation efficiency. As having developed the simulation code, High-level rAdiowaste Disposal Evaluation System (HADES), to describe TH behavior based on Multi-physics Object-Oriented Simulation Environment (MOOSE) software, the authors make a milestone to develop and couple the MOOSE-based new code for M behavior as Sub-app, with the previous HADES set to be Main-app. New model for M behavior will be verified with the benchmark case of DECOVALEX-THMC Task D, comparing the mechanical simulation results: stress evolution over time, profiles of stress and vertical displacement. The existing simulation results from HADES will also be updated with the coupled calculations, with regard to temperature and saturation. Additionally, the effective stress evolution can be assessed in terms of repository's stability with Spalling Strength and Mohr-Coulomb failure criterion. This concept for new simulation model has its meaning in that it aims to demonstrate the specific methodology of loosely coupling multi-physics in unified simulation code and analyze THM complex interactions with considering mutual influence on various physics. It is expected that HADES can be renewed as an integral simulation model for deep geological repository systems by possessing the capacity for analyzing and assessing mechanical behavior.

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Keywords: Deep geological repository, High-level radioactive waste, 3D coupled THM model, Numerical simulation, MOOSE

Expert Opinion Collection Methodology and Integrated FEP List Development for High-level Radioactive Waste Disposal Site Establishment

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Spent nuclear fuel temporary storage in South Korea is approximately 70% of total storage capacity as of the 4th quarter of 2022 amount is stored. In addition, 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.

Globally, after the conclusion of the EU Taxonomy, for nuclear energy in order to become an eco-friendly energy, it is necessary to have a high-level radioactive waste disposal site and submit a detailed operation plan for high-level radioactive waste disposal site by 2050. Finland and Sweden have already received permission for the construction of high-level radioactive waste disposal facilities, and other countries, such as Switzerland, Japan, the United States, and Canada, are in the process of licensing disposal facilities.

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. Since it is a characteristic of the Systematic method to compose a scenario by combining these FEP, the Systematic method is the basic premise for the development of FEP.

Completeness is important for FEP, and comprehensiveness is important for scenarios. However, combining all the FEP into one scenario is time-consuming and difficult to ascertain the comprehensiveness of the scenario. Therefore, an Integrated FEP list is being developed to facilitate tracking between FEP and scenarios by integrating similar FEP.

In this study, during the integrated FEP development process, a method for utilizing experts that can be used for difficult parts of quantitative evaluation and a quantitative evaluation process through the method were presented.

Keywords: High-level radioactive waste, Disposal, Safety case, FEP, Expert opinion

A Review on How to Assess the Accessory Mineral Alteration in Bentonite in the Domestic Disposal Facility for High-level Radioactive Waste

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For the deep geological repository, engineering barrier system (EBS) is installed to restrict a release of radionuclide, groundwater infiltration, and unintentional human intrusion. Bentonite, mainly used as buffer and backfill materials, is composed of smectite and accessory minerals (e.g. salts, silica). During the post-closure phase, accessory minerals of bentonite may be redistributed through dissolution and precipitation due to thermal-hydraulic gradient formed by decay heat of spent nuclear fuel and groundwater inflow. It should be considered important since this cause canister corrosion and bentonite cementation, which consequently affect a performance of EBS. Accordingly, in this study, we first reviewed the analyses for the phenomenon carried out as part of construction permit and/or operating license applications in Sweden and Finland, and then summarized the prerequisite necessary to apply to the domestic disposal facility in the future.

In previous studies in Sweden (SKB) and Finland (POSIVA), the accessory mineral alteration for the post-closure period was evaluated using TOUGHREACT, a kind of thermal-hydro-geochemical code. As a result of both analyses, it was found that anhydrite and calcite were precipitated at the canister surface, but the amount of calcite precipitate was insignificant. In addition, it was observed that precipitate of silica was negligible in POSIVA and there was a change in bentonite porosity due to precipitation of salts in SKB.

Under the deep disposal conditions, the alteration of accessory minerals may have a meaningful influence on performance of the canister and buffer. However, for the backfill and closure, this is expected to be insignificant in that the thermal-hydraulic gradient inducing the alteration is low. As a result, for the performance assessment of domestic disposal facility, it is confirmed that a study on the alteration of accessory minerals in buffer bentonite is first required. However, in the study, the following data should reflect the domestic-specific characteristics: (a) detailed geometry of canister and buffer, (b) thermal and physical properties of canister, bentonite and host-rock in the disposal site, (c) geochemical parameters of bentonite, (d) initial composition of minerals and porewater in bentonite, (e) groundwater composition, and (f) decay heat of spent nuclear fuel in canister.

It is presumed that insights from case studies for the accessory mineral alteration could be directly applied to the design and performance assessment of EBS, provided that input data specific to the domestic disposal facility is prepared for the assessment required.

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Keywords: Bentonite alteration, Performance assessment, Thermal-hydraulic-geochemical analyses, Post-closure performance

~~Numerical Analysis of the Barcelona Basic Model by the Discontinuum-based Method~~

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Withdrawal

Manufacturing of Real-scale CANDU Disposal Canister

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In-depth disposal of spent nuclear fuel means safe disposal of spent nuclear fuel by the concept of a multi-barrier system composed of an artificial barrier, an engineering barrier, and a natural barrier system of natural rock at a depth of less than 500 m underground. Disposal canisters are needed to store high-level waste in a deep environmental for a long time, and in order to demonstrate the performance of deep disposal canisters for spent nuclear fuel at underground research facilities (URL), it is intended to design disposal canisters and manufacture internal canisters.

The internal canisters of spent nuclear fuel disposal canisters manufactured as a result of the study are combined with external copper canister technology and are directly used for demonstration of engineering barrier performance in underground facilities (URL) essential for final disposal of spent nuclear fuel. Disposal canister manufacturing technology and manufacturing process are used to manufacture disposal canisters for future final disposal projects in connection with domestic unique disposal systems. The quality inspection and quality management technology applied when manufacturing disposal canisters contribute to securing the soundness of disposal canisters that primarily maintain the safety of in-depth disposal by using them in the actual disposal business. By visually showing the development status of domestic disposal technology by displaying the prototype of disposal canisters manufactured as major achievements, the public can raise awareness of the domestic technology and safety of in-depth disposal of spent nuclear fuel.

Keywords: Disposal canister, CANDU, Spent nuclear fuel, Deep geological disposal

Training Data Production for Developing Machine Learning Based Hybrid Solver for Disposal Repository

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To conduct numerical simulation of a disposal repository of the spent nuclear fuel, it is necessary to numerically simulate the entire domain, which is composed of numerous finite elements, for at least several tens of thousands of years. This approach presents a significant computational challenge, as obtaining solutions through the numerical simulation for the entire domain is not a straightforward task.

To overcome this challenge, this study presents the process of producing the training data set required for developing the machine learning based hybrid solver. The hybrid solver is designed to correct results of the numerical simulation composed of coarse elements to the finer elements which derive more accurate and precise results. When the machine learning based hybrid solver is used, it is expected to have a computational efficiency more than 10 times higher than the numerical simulation composed of fine elements with similar accuracy.

This study aims to investigate the usefulness of generating the training data set required for the development of the hybrid solver for disposal repository. The development of the hybrid solver will provide a more efficient and effective approach for analyzing disposal repository, which will be of great importance for ensuring the safe and effective disposal of the spent nuclear fuel.

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Keywords: Numerical simulation, Disposal repository, Machine-learning, Hybrid solver

Effect of Interfacial Properties on Numerical Modeling of Gas Migration in Field-Scale Engineered Barrier Systems

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The engineered barrier system (EBS) is an indispensable element of a deep geological repository (DGR) designed to prevent the discharge of radioactive materials into the environment. The buffer material is a vital component of the EBS by creating a physical and chemical barrier that prevents the migration of radioactive materials. In the disposal environment, gases can be generated from the corrosion of the canister. When the gas generation rate exceeds the diffusion rate, the buffer material's performance can deteriorate by the physical damage induced by the increase in pore pressure. Therefore, understanding the EBS's behavior under gas generation conditions is crucial to guarantee the long-term safety and performance of the DGR. Lab-scale and field-scale experiments have been conducted to examine the stability of the buffer material concerning gas generation and movement by the previous researchers. To evaluate long-term stability for more than 100,000 years, it is essential to assess stability using a numerical model verified by these experiments. This study investigated the effect of interfacial characteristics on the numerical modeling accuracy of experimental simulation while verifying a numerical model through field-scale experimental results. The findings of this study are expected to furnish fundamental data for establishing numerical analysis guidelines for the long-term stability assessment of disposal systems.

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Keywords: Engineered barrier system, Numerical verification, Field-scale test, Interface characteristics, Gas injection test

Comparison of Evapotranspiration Estimated Using Simple and Complex Models

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Surface environmental factors such as climate change can affect the safety of the disposal system by changing groundwater recharge or flow. Therefore, it is important to identify surface environmental factors and hydrogeological factors to evaluate long-term changes in hydrogeological environment of a disposal system. In particular, evapotranspiration is an important to be considered because it loses 70% of rainfall and has a great effect on groundwater recharge.

Evapotranspiration can be estimated using simple or complex models based on meteorological data. Meteorological data from January 2010 to December 2022 were collected from 44 Automatic Synoptic Observation Systems (ASOS) of the Korea Meteorological Administration (KMA), which observe factors necessary for calculating evapotranspiration. For the estimation of evapotranspiration through simple models, temperature-based models (Blaney-Criddle method, modified Blaney-Criddle method, Hargreaves-Samani method) and radiation-based models (Simple Abtew method, Makkink method, Priestley-Taylor method, Turc method, Solar radiation-Maximum temperature method) were used. The calculation of evapotranspiration through the complex model used the Penman-Monteith method, which is used as a standard model in the USA, Japan, and FAO. By comparing the evapotranspiration calculated by complex and simple model, methods with small errors were identified each region. In addition, long-term climate change scenarios were applied to confirm changes in long-term evapotranspiration in South Korea. The results of this study will be used to find alternative models in the case of missing data in the Penman-Monteith model, which requires a lot of meteorological data, and can be used as basic data for calculating groundwater recharge that can affect the disposal system in the future.

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Keywords: Evapotranspiration, Radiation-based methods, Temperature-based methods, Penman-monteith model

The Establishment of Safety Case Analysis Database for Deep Geological Disposal System in Leading Foreign Countries Using Artificial Intelligence

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Since high-level radioactive wastes contain long-lived nuclides and emit high energy, they should be disposed of permanently through a deep geological disposal system. In Korea, the first (2016.07) and the second (2021.12) basic plans for the management of high-level disposal systems were proposed to select sites for deep geological disposal facilities and to implement business strategies.

Leading countries such as Finland, Sweden and France have developed and applied safety cases to verify the safety of deep geological disposal systems. By examining the regulatory status of foreign leading countries, we analyze the safety cases ranging from the site selection stage of the deep geological disposal system to the securing of the permanent disposal system to the investigation, analysis, evaluation, design, construction, operation, and closure. Based on this analysis, we will develop safety case elements for long-term safety of deep geological disposal systems suitable for domestic situation.

To systemically analyze data based on safety cases, we have established a database of deep geological disposal system regulations in leading foreign countries. Artificial intelligence text mining and data visualization techniques are used to provide database in dashboard form rather than simple lists of data items, which is a limitation of existing methods. This allows regulatory developers to understand information more quickly and intuitively and provide a convenient interface so that anyone can easily access the analyzed data and create meaningful information. Furthermore, based on the accumulated bigdata, the artificial intelligence learns and analyzes the information in the database through deep learning, and aims to derive a more accurate safety case.

Based on these technologies, this study analyzed the legal systems, regulatory standards, and cases of major international leading countries and international organizations such as the United States, Sweden, Finland, Canada, Switzerland, and the IAEA to establish a database management system. To establish a safety regulation base suitable for the domestic deep geological disposal environment, the database is provided as data to refer to and apply systematic information management on regulatory standards and regulatory cases of overseas leading countries, and it is expected that it will play a key role as a forum for understanding and discussing the level of safety of deep geological disposal system among stakeholders.

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Keywords: Deep geological disposal system, High-level radioactive wastes, Database, Regulatory development, Data analysis

Predicting Hydrologic Response to Climate Changes for Safety Assessment of Geological Disposal System

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The timescale of safety assessment for a geological disposal system is considered up to hundreds of thousands of years when the radionuclides in spent nuclear fuel decay to levels comparable to natural radioactivity. During this long period, a variety of climate changes are expected to occur, including variations in temperature and precipitation as well as long-term sea level changes and glacial cycles. These climate changes can either directly affect water balance components or indirectly affect water balance by altering terrain and vegetation that have an impact on water balance. Water balance is a significant element of safety assessment, because it affects the radionuclide transport via groundwater flow, which in turn affects the radiological risk to humans and other biotas. Therefore, it is important to understand the hydrologic response to climate changes for proving the long-term safety of the disposal system.

To this end, this study performed hydrological simulations using the SWAT (Soil and Water Assessment Tool) for several climate change scenarios. SWAT is the watershed-scale hydrological model developed by the USDA-ARS (United States Department of Agriculture - Agricultural Research Service) and has been widely used to quantify the water balance in a watershed. It calculates the hydrologic cycle based on the water balance equation with different physical processes for water balance components such as evapotranspiration, surface runoff, and groundwater recharge. This study assumed several climate change scenarios (e.g., variations in temperature and precipitation, sea level change, and formation of permafrost) and analyzed how the components of the water balance would respond under different scenarios and which scenarios would have the greatest impact on the water balance. These findings can provide valuable insights for future long-term safety assessments on the Korean Peninsula and can also be used as input data for the biosphere module of APro (Adaptive process-based total system performance assessment framework).

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Keywords: Climate change, Water balance, SWAT, Hydrological model, Safety assessment, Geological disposal system

Characterization of Geometric Properties of Rough Rock Joints for Hydromechanical Analyses

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Discontinuities exert great influence on the thermal, hydraulic, and mechanical behavior of rock mass. Rock joint is one of the most frequently encountered discontinuities in many engineering applications, such as tunnel, rock slope and repository for high level radioactive waste. Therefore, the effects of rock joint should be thoroughly investigated in various aspects. Rock joint has gone through many geological processes and its behavior can be characterized by many properties. Among them, geometric properties, such as joint roughness, aperture, and contact area can affect mechanical and hydraulic properties and vice versa. Therefore, accurate understanding and characterization of the geometric properties are of importance. Generally, the geometric properties of a joint are obtained or estimated using the surface height or elevation, which could be measured by various contact or non-contact methods. Then, the coordinates of the surfaces are used to calculate several parameters, for instance roughness indexes and mechanical aperture, in a quantitative manner.

This paper is a part of SKB task force project that aims to evaluate the geometric properties of rock joints and to analyze the hydromechanical behavior within a rough joint considering the properties. Four pairs of joint surfaces were laser-scanned in order to obtain coordinates of the surfaces and then the coordinates were used to calculate the roughness, directional roughness, aperture, and spatial correlations. At the same time, fluid flow within a rough joint were simulated by a commercial FEM code, considering the variation of aperture space due to normal load. Flowrate, flow path, and channelization were investigated in an aperture scale. Since rock mass consists of several joints and/or joint sets, characterization of a single rock joint can be utilized for analyzing the behavior of rock mass as a reference.

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Keywords: Rock joint, Roughness, Aperture, Fluid flow, Numerical simulation

Preliminary Study on Wireless Communication Between Compacted Bentonite Blocks

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High level radioactive waste disposal repository is faced thermos-hydro-mechanical-radioactive condition. Factors according to these complex conditions are measured using multiple sensors installed in the disposal repository to check integrity of the structure. Wires of the sensors can be potential pathways of groundwater and nuclide flow and these pathways accelerates bentonite saturation. Therefore, it is worth to developing wireless sensors buried in the bentonite buffer which can communicate without wires. In start of the study, widely-utilized wireless communication methods including WiFi and LoRa are tested using compacted bentonite blocks to estimate the performance of them. Compacted bentonite blocks are prepared using di-press method with metal molds and the dry density of them are about 1.6 g/cm³. All wireless communication methods are well communicated through the bentonite blocks over 50 cm. The further experimental tests will be conducted with different dry density and water contents. The results of these experimental tests give a possibility of wireless communications in compacted bentonite buffer and will be utilized for the design of wireless sensor systems for the repository monitoring.

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Keywords: Bentonite buffer, Wireless sensors, WiFi, LoRa

The Preliminary Test of the Magnetic Fabric of Fault Gouge From KURT

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A radioactive waste repository consists of engineered barriers and natural barriers and must be safely managed after isolation. Geologic events in natural barriers should be categorized and evaluated according to their magnitude to assess the present and future stability of disposal. Among the long-term evolutionary elements of natural barriers, faults are a small portion of the Earth's crust. Still, they play an important role in nuclide transport as conduits for fluids moving deep underground. In addition, the physical and chemical properties of fault rocks are useful for understanding the long-term and short-term behavior of faults.

Paleomagnetic research has been used extensively and successfully for igneous, metamorphic, and sedimentary rocks. In addition, magnetic characterization of fault rocks can be used to describe faults or infer the timing of major geological events along fault zones. Components of magnetization defined in fault-breccias were attributed to chemical processes associated with hydrothermal mineralization that accompanied or post-dated tectonic activity along the fault. The study of magnetic minerals in fault rocks can be used as "strain indicators", "geothermometers", etc.

This study is a preliminary test of magnetic properties using fault gouges. Fault gouges are not well preserved in typical terrestrial environments. Access to fresh gouges typically requires trenching through faults or sampling with a core drill. Fortunately, it is a magnetic property study using a fault gouge that exists on the inner wall of KURT (KAERI Underground Research Tunnel). This is to identify the motion history of the fault and, furthermore, to understand the stress structure at the time of fault creation. In addition, it can be presented as evidence for evaluating faults that may appear in future URL (Underground Research Laboratory).

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Keywords: Natural barrier, Geological history, Rock and mineral magnetism, Paleomagnetism

Estimation of Soil Water Balance Using Weighing Lysimeter Around KAERI Underground Research Tunnel

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Long-term evolution of the surface environments can affect the safety of deep geological disposal. Therefore, it is important to understand the water balance components constituting the water cycle among atmosphere, surface, and subsurface. In Finland, the surface and near-surface hydrological model (SHYD) was developed to calculate the water balance of Olkiluoto Island. Through the intensive site investigations, the data sets as input for the site scale model in present-day conditions have been collected such as transpiration and meteorological data. In this study, weighing lysimeter method was selected to quantify small-scale soil water balance of the vadose zone in the Unsaturated zone In-situ Test facility (UNIT) around KAERI Underground Research Tunnel. Hydrological components such as precipitation, evapotranspiration (ET) and leachate were derived from water balance analysis on the lysimeter measurements in UNIT. Among the hydrological components, actual ET accounts for more than 50% of the annual precipitation, and thus plays an important role on predicting the hydrological evolution in the future. In this context, actual ET measured from the weighing lysimeter was compared with potential ET estimated from meteorological data using FAO-56 Penman-Monteith method.

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Keywords: Safety, Water cycle, Hydrological component, Evapotranspiration, KURT

Evaluation of Aperture Distribution in Fractured Granite Block Using CFD Analysis

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A deep geological repository for disposal of high-level radioactive waste (HLW) consists of the canister, buffer material, and natural rock. If radionuclides leak from a disposal container, it can pass through buffer materials and rock, and move into the biosphere. Transport and migration of radionuclides in the rock differently were affected by the fracture type, filling minerals in the fracture, and the chemical and hydraulic properties of the groundwater. In this study, aperture distribution in fractured granite block was investigated by hydraulic test and CFD analysis. The fractured rock block ($1\text{ m} \times 0.6\text{ m} \times 0.6\text{ m}$), which is simulated as natural barrier, was prepared from Iksan, Jeollabuk-do. 9 test holes were drilled and packer system was installed to perform hydraulic test at the surface of fracture. 3D model simulated for aperture distribution of rock block was made using results of hydraulic test. And then, CFD analysis was performed to evaluate the co-relation between experiment results and analysis results using FLUENT code.

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Keywords: Fractured rock, Granite, Aperture distribution, Computational fluid dynamics (CFD)

Development of Bentonite Buffer Degradation Model Due to the Intrusion Into the Discontinuity Near the Deposition Hole

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Bentonite is a widely used buffer material in high-level radioactive waste repositories due to its favorable properties, including its ability to swell and low permeability. Bentonite buffers play an important role in safe disposal by providing a low permeability barrier and preventing radionuclides migration into the surrounding rock. However, the long-term performance of the bentonite buffer is still an area of research, and one of the main concerns is the erosion of the buffer due to swelling and groundwater flow. Erosion of the bentonite buffer can have a significant impact on repository safety by reducing the integrity of the buffer and forming colloids that can transport radionuclides through groundwater, potentially increasing the risk of radionuclide migration. Therefore, understanding the mechanisms and factors that influence the erosion of the bentonite buffer is critical to the safety assessment of high-level radioactive waste repositories.

In this study, we attempted to develop the bentonite buffer erosion model using Adaptive Process-based total system performance assessment framework for a geological disposal system (APro) proposed by the Korea Atomic Energy Research Institute (KAERI). First, the erosion phenomenon was divided into two stages: bentonite buffer penetration into rock fractures and colloid formation. As an initial step in the development of the buffer erosion model, a bentonite buffer intrusion into the fracture and consequent degradation of buffer property were considered. For this purpose, a two-region model based on the dynamic bentonite diffusion model was adopted which is one of the methods for simulating bentonite buffer intrusion. And, it was assumed that the buffer properties, such as density, porosity and permeability, thermal conductivity, modulus of elasticity, and mechanical strength, are degraded as the buffer erodes. The bentonite buffer degradation model developed in this study will serve as a foundation for the comprehensive buffer erosion model, in conjunction with the colloidal formation model in the future.

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Keywords: High-level radioactive waste repository, Process-based total system performance assessment, APro, Bentonite buffer erosion, Dynamic bentonite diffusion model, Two-region model

Chemical Erosion and Colloidal Release of Calcium Bentonite in an Artificial Fracture System

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Bentonite is a promising buffer material for high-level radioactive waste (HLW) disposal due to the high nuclides sorption capacity and swelling property. However, bentonite has the potential to generate colloid particles, with small particle sizes less than 1,000 nm when in contact with groundwater. The bentonite colloids easily form pseudo-colloid with the released nuclides and migrate through the water-conducting rock to the biosphere. Therefore, understanding the generation and migration of bentonite colloids is crucial in assessing the safety of the HLW repository. In this study, an artificial fracture system was prepared to investigate colloid release from compacted bentonite. A 250 mm diameter acrylic artificial fracture system was used, with 30 mm of compacted calcium bentonite installed. Artificial groundwater flow was injected into the system at a flow rate of 250 $\mu\text{L}/\text{h}$, and every 6 mL of leachate was collected by a fraction collector. A film-type pressure sensor was equipped to monitor the swelling pressure, and the swelling was observed using a digital microscope. The results indicate that the compacted bentonite formed a mineral ring originating from the swelling of the bentonite, and the end of the ring generated colloid particles due to chemical erosion. Although the release rate of colloids increased with increasing flow rate, the colloid ratio depended on the low ionic strength of the injected artificial groundwater. This work contributes to the understanding of the chemical erosion and colloid release mechanism of compacted bentonite.

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Keywords: Colloid, Nuclide, Compacted bentonite, Chemical erosion, Ionic strength, Size distribution, Deep geological repository

Verification of Accuracy and Calculation Efficiency of Global-Local Iterative Analysis Method for Large-scale Process-based TSPA

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The most important thing in development of a process-based TSPA (Total System Performance Assessment) tool for large-scale disposal systems (like APro) is to use efficient numerical analysis methods for the large-scale problems. When analyzing the borehole in which the most diverse physical phenomena occur in connection with each other, the finest mesh in the system is applied to increase the analysis accuracy. Since thousands of such boreholes would be placed in the future disposal system, the numerical analysis for the system becomes significantly slower, or even impossible due to the memory problem in cases.

In this study, we propose a tractable approach, so called global-local iterative analysis method, to solve the large-scale process-based TSPA problem numerically. The global-local iterative analysis method goes through the following process: 1) By applying a coarse mesh to the borehole area the size of the problem of global domain (entire disposal system) is reduced and the numerical analysis is performed for the global domain. 2) Solutions in previous step are used as a boundary condition of the problem of local domain (a unit space containing one borehole and little part of rock), the fine mesh is applied to the borehole area, and the numerical analysis is performed for each local domain. 3) Solutions in previous step are used as boundary conditions of boreholes in the problem of global domain and the numerical analysis is performed for the global domain. 4) steps 2) and 3) are repeated.

The solution derived by the global-local iterative analysis method is expected to be closer to the solution derived by the numerical analysis of the global problem applying the fine mesh to boreholes. In addition, since local problems become independent problems the parallel computing can be introduced to increase calculation efficiency. This study analyzes the numerical error of the global-local iterative analysis method and evaluates the number of iterations in which the solution satisfies the convergence criteria. And increasing computational efficiency from the parallel computing using HPC system is also analyzed.

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Keywords: APro, Large-scale process-based TSPA, Global-local iterative analysis method, Numerical accuracy, Parallel computation performance

A Method for Combination of Top-down and Bottom-up Approaches for Scenario Development

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To prove the long-term safety of deep geological repository, the safety assessment is needed to ensure that the expected performance of repository satisfies the regulatory standards. Scenario development is process of analyzing events and evolutions that can directly or indirectly affect the performance of a disposal system and is a pre-step for quantitative safety assessment. Scenarios are used to identify and define cases to be assessed by numerical modeling, and cases are mainly divided into normal (also called the ‘reference’ and ‘expected evolution’) and abnormal scenarios. Mainly two approaches have been used to set up scenarios. One is a bottom-up approach that starts with features, events and processes (FEPs). This approach can analyze the evolution and events related to the performance of the disposal system in an inductive manner. The other is top-down approach that analyzes the events and evolution of disposal system, focusing on situations that may affect the safety function of the components. This approach starts with a set of intuitively predefined expected failures of safety function. Combining the two approaches is more effective in demonstrating comprehensiveness which is a main challenge of scenario analysis, and almost national radioactive waste management institutions combine top-down and bottom-up approaches for development of scenarios. An approach combining the two approaches is called a hybrid approach, and the detailed method differs from each institution and has not been determined. In this study, some work for constructing the scenario using hybrid approach was performed. Firstly, defining each component’s safety function and screening FEPs according to several rules were performed for a generic repository. Secondly, we extracted performance factors that are considered likely to affect safety functions. And lastly, we integrated FEPs correlated with performance factor to simplify the analysis. These results will be material to construct the scenario using hybrid approach.

Acknowledgements

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Keywords: Safety assessment, Scenario, FEP, Deep geological repository, Safety case

A Preliminary Study on the Development of Methodology for Predicting Future Geological Evolution Affecting the Long-term Safety of Disposal Site

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To prevent the release of radionuclides into the biosphere, disposal facilities for radioactive waste should be located to provide isolation from the accessible biosphere for tens of thousands to a million years after closure. During the period of interest, the constantly evolving natural environment and possible geological events of the site can cause disturbances to the containment function of the repository. Thus, for the long-term safety assessment of the repository, the possible long-term change of natural barrier should be considered. Due to the characteristics of radionuclides that transport mainly through the groundwater, understanding the long-term evolution of groundwater flow and geochemical properties is essential to assess the long-term changes in the natural barrier performance. The changes in characteristics of natural rocks and geological structures are one of the main factors that determine the hydrological and geochemical characteristics of the deep underground. In this study, we plan to develop a methodology to estimate these future geological evolutions in order to assess the possibility of hazardous events of the site that can affect hydrological or geochemical properties over the period of interest, and also in order to verify the change in the geological environment is within the safe performance range even after the period of interest. However, it is very unreliable to predict future changes in the natural environment because it is very heterogeneous, complex, and difficult to observe directly. For the preliminary study of the project, we reviewed cases of future evolution prediction researches with regard to the geological environment of disposal site and methods they applied to reduce the uncertainty of the prediction. The results will be used to establish basic data for future studies on the long-term evolution of hydraulic-mechanics performance of natural barrier and long-term evolution of geochemical performance around KURT site. In addition, it can contribute to construct long-term evolution scenario of the geological environment around future URL site.

Acknowledgements

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Keywords: Geological disposal, Disposal site, Long-term safety, Geological evolution

Assessment of Compressive Wave Velocity and Thermal-mechanical Property of Compacted Bentonite Buffer Blocks for Radioactive Waste Disposal Repository

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The engineered barrier system (EBS) for deep geological disposal of high-level radioactive waste requires a buffer material that can prevent groundwater infiltration, protect the canister, dissipate decay heat effectively, and delay the transport of radioactive materials. To meet those stringent performance criteria, the buffer material is prepared as a compacted block with high-density using various press methods. However, crack and degradation induced by stress relaxation and moisture changes in the compacted bentonite blocks, which are manufactured according to the geometry of the disposal hole, can critically affect the performance of the buffer. Therefore, it is imperative to develop an adequate method for quality assessment of the compacted buffer block. Recently, several non-destructive testing methods, including elastic wave measurement technology, have been attempted to evaluate the quality and aging of various construction materials. In this study, we have evaluated the compressive wave velocity of compacted bentonite blocks via the ultrasonic velocity method (UVM) and free-free resonant column method (FFRC), and analyzed the relationship among compressive wave velocity, dry density, thermal conductivity, and strength parameter. We prepared compacted bentonite block specimens using the cold isostatic pressure (CIP) method under different water content and CIP pressure conditions. Based on multiple regression analysis, we suggest a prediction model for dry density in terms of manufacturing conditions. Additionally, we propose an empirical model to predict thermal conductivity and unconfined compressive strength based on compressive wave velocity. The database and suggested models in this study can contribute to the development of quality assessment and prediction techniques for compacted buffer blocks used in the construction of a disposal repository.

Acknowledgements

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Keywords: Compacted bentonite block, Quality assessment, Elastic wave velocity, Thermal conductivity, Unconfined compressive strength

Fault Activity Direction and Age Estimation for Safety Evaluation of Radioactive Waste Facilities and Nuclear Power Plant Sites

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Fault activity acts as the greatest risk factor in relation to the stability of the radioactive waste disposal facilities and nuclear power plant site, and for this reason, geological studies on areas with past fault activity history must precede site evaluation studies. This study aims to trace the fault activity history of large fault zones, including the Yangsan fault in the southeastern part of the Korean Peninsula, where two major earthquakes occurred, and to obtain fault activity direction information that is the basis for stability evaluation. The 3D-Shape Preferred Orientation (SPO) of particles in the fault rock created by the earthquake was investigated to analyze the direction of fault plane activity, and the age of fault activity was estimated through Illite Age Analysis (IAA) analysis. It is expected that the large-scale fault activity information in the southeastern part of the Korean Peninsula obtained through the SPO and IAA analysis can be used as basic data for safety evaluation of existing or future nuclear power plants and radioactive waste facilities.

Keywords: Radioactive waste facilities, Nuclear power plants, Safety evaluation, Shape preferred orientation (SPO), Illite age analysis (IAA), Fault activity

4분과

중저준위폐기물관리(Oral)

Review of Compliance With the Waste Acceptance Criteria of Paraffin-Solidifying Concentrate Waste Packaged Into HIC for Permanent Disposal

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Domestic NPPs had produced the paraffin-solidifying concentrate waste (PSCW) for nearly 20 years. At that time radioactive waste management policy of KHNP was to reduce the volume and to store safely in site. The PSCW has been identified not to meet the leaching index after introducing the treatment system. PSCW has to be treated to meet current waste acceptance criteria (WAC) for permanent disposal.

PSCW consists of dried concentrate 75% and paraffin 25% of volume. When PSCW is separated into a dried concentrate and a paraffin by solubility, total volume separated is increased twice. Final disposal volume of dried concentrate can reach to several times when solidifying by cement even considering exemption. Application of polymer solidification technology is difficult because dried concentrate is hard to make form to pellet. When PSCW is packaged in High Integrity Container (HIC), volume of PSCW is equal to the volume before package. The packaging process of HIC is simple and is no necessary of large equipment.

It is important to recognize that HIC was developed to replace solidification of waste. HIC has as design goal a minimum lifetime of 300 years under disposal environment. The HIC is designed to maintain its structural integrity over this period, to consider the corrosive and chemical effects of both the waste contents and the disposal environment, to have sufficient mechanical strength to withstand loads on the container and to be capable of meeting the requirements for a Type A transport Package.

The Final waste form is required for facilitating handling and providing protection of personnel in relation to solidification, explosive decomposition, toxic gases, hazardous material, etc.

Structural stability of final waste form is required also. Structural stability of the waste can be provided by the waste itself, solidifying or placing in HIC. Final waste form ensure that the waste does not structurally degrade and affect overall stability of the disposal site.

The HIC package contained PSCW was reviewed from several points of view such as physicochemical, radiological and structural safety according to domestic WAC. The result of reviewing shows that it has not found any violation of WCP established for silo type disposal facility in Gyeongju city.

Keywords: Waste concentrates, Paraffin solidification, Waste acceptance criteria, LILW, Disposal

Effects of Bentonite Erosion and Intrusion on the Mechanical Properties of Jointed Rock Mass

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Compacted bentonite buffer blocks placed in the engineered barrier system for high-level nuclear waste disposal can undergo swelling, intrusion into rock fractures, and erosion with saturation. Bentonite erosion and intrusion can lead to bentonite mass loss via groundwater flow and can ultimately compromise the overall integrity of the disposal system. To ensure the long-term safety of deep geological disposal, it is essential to assess the hydro-mechanical interactions between the bentonite buffer and surrounding rock.

In this study, the impact of bentonite erosion and intrusion on the mechanical properties of the jointed rock mass were assessed via elastic wave propagation measurements using the quasi-static resonant column test. Granite rock discs obtained from the Korea Underground Research Tunnel and Gyeongju bentonite were used to simulate jointed rock specimens with different bentonite intrusion conditions. Different degrees of bentonite intrusion were simulated by mixing bentonite and water to create bentonite paste and gel. The longitudinal and shear wave velocities under different normal stress levels were used to quantify the effects of bentonite intrusion on the mechanical characteristics of the rock joint. Complementary numerical analysis using the three-dimensional distinct element code (3DEC) was conducted to provide improved understanding of wave propagation within bentonite gouge-filled rock mass.

Acknowledgements

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Keywords: Engineered barrier system, Bentonite erosion and intrusion, Jointed rock mass, Long-wavelength Elastic wave velocity, Quasi-static resonant column test, 3DEC

Selective Treatment of C-14 in Spent Ion Exchange Resin From HWR Using the Microwave

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Mixed-bed ion exchange resin consist of anion exchange resin and cation exchange resin is used to treat liquid radioactive waste in nuclear power plants. C-14 from heavy water reactors (HWR) is adsorbed on the anion exchange resin and is considered intermediate-level radioactive waste. The total amount of radioactivity of C-14 in spent ion exchange resin exceeds the activity limits for the disposal facility. Therefore, it is necessary to reduce the radioactivity through pre-treatment. There are thermal and non-thermal methods for the treatment of spent ion exchange resin. However, destructive methods have the problem of emitting off-gas containing radionuclides. To solve this challenge, various methods have been developed such as acid stripping, PLO process, activity stripping, thermal treatment and others.

In this study, spent ion exchange resin (spent resin) was treated using microwave. The reaction characteristics of the resin to microwave were used to selectively remove the C-14 on the functional groups. Simulated spent anion exchange resin and spent resin from Wolsong NPP were treated with the microwave method, and the desorption rate was over 95%. An integrated process system of 1 kg/batch was built to produce operating data. After the operation of the process, characterization and evaluation of post-treatment for condensate water and adsorbent used in the process were performed. When the process system was applied to treat simulated spent resin and real spent resin, both showed a desorption rated of more than 97%. It means that the C-14 was successfully removed from the radioactive spent resin.

Keywords: Spent ion-exchange resin, C-14, Microwave, Intermediated level radioactive waste

Self-attenuation Correction of U-235 in Metal Ingot Using MCNP and Gamma Spectroscopy

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Some of the metal waste generated from KEPCO NF is being disposed of in the form of ingots. An ingot is a metal that is melted once and then poured into a mold to harden, and it is characterized by a uniform distribution of radioactive material. When measuring the uranium radioactivity in metal ingot with HPGe detector, 185.7 keV of U-235 is used typically because most gamma rays emitted at U-235 are distributed in low-energy regions below 200 keV. To analyze radioactivity concentration of U-235 with HPGe detector more accurately, self-attenuation due to geometrical differences between the calibration source and the sample must be corrected.

In this study, the MCNP code was used to simulate the HPGe gamma spectroscopy system, and various processes were performed to prove the correlation with the actual values. First an metal ingot-type standard source was manufactured for efficiency calibration, and the GEB coefficient was derived using Origin program. And through the comparison of actual measurements and simulations, the thickness of the detector's dead layers were defined in all directions of Ge crystal. Additionally instead of making an metal ingot-type standard source every time, we analyzed the measurement tendency between commercially available HPGe calibration source (Marinelli beaker type) and the sample (metal ingot type), and derived the correction factor for geometry differences. Lastly the correction factor was taken into consideration when obtaining the uranium radioactivity concentration in the metal ingot with HPGe gamma spectroscopy.

In conclusion, the U-235 radioactivity in metal ingot was underestimated about 25% of content due to the self-attenuation. Therefore it is reasonable to reflect this correction factor in the calculation of U-235 radioactivity concentration.

Keywords: Self-attenuation correction, Ingot, Gamma spectroscopy

Leaching Behavior of Cs, Sr, and Co Inside the Cement Solidified Specimen According to the Leachant Type

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Several tests should be performed to estimate the structural and chemical stability of the radioactive waste. Among the tests in Gyeongju LILW repository, the leaching test which follows ANS 16.1 standard test method should be conducted for Cs, Sr, and Co radionuclides and must satisfy leachability index larger than 6 which applies deionized water as a leachant. However, the expected leachant inside the silo is groundwater that contains various ions and a high pH condition is predicted due to the concrete structures inside the silo.

According to the chemical environment of the leachant, the chemical form of the radionuclides varies from precipitate to ion. Cobalt precipitates when the leachant has high pH environment which is similar condition to the cement-saturated leachant. Unlike the cobalt, cesium is preferred to exist as ion in the high pH condition. Therefore, the significant effect of the chemical environment of the leachant on the leachability of the radionuclides should be considered for the waste acceptance criteria of the radioactive waste repository.

From the 'NRC, Technical position on the waste form, rev1', the leaching test method should follow the ANS 16.1 methods by using deionized water as leachant, however, a new leachant showing more aggressive leachability can be applied instead of deionized water. In the other hand, ASTM C1308 leaching test method recommends applying actual groundwater of the repository as a leachant. FT-04-020, the leaching test method of France, suggests the ion composition of the groundwater including the pH value. Therefore, the adequacy of using deionized water as leachant for the leaching test method of Cs, Sr, and Co should be re-examined.

In this study, the leaching behavior of Cs, Sr, and Co under the several leachant types is estimated. The cement solidified specimen containing single Cs, Sr, and Co were manufactured. The leaching test following ANS 16.1 was performed by applying deionized water, simulated groundwater, and cement-saturated groundwater. As a result, a leachability index difference according to the leachant type was discussed. The result of this study is expected to be a background data that helps understanding the actual leaching behavior of the Cs, Sr, and Co in the Gyeongju LILW repository.

Acknowledgements

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Keywords: Leachant, Leaching behavior, Leachability index, Radioactive waste, Repository

Analysis on Factors to Be Considered for the Safety Assessment of Colloids

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Concerns with colloids, dispersed 1~1,000 nm particles, in the LILW repository are being raised due to their potential to enhance radionuclide release. Due to their large surface areas, radionuclides may sorb onto mobile colloids, and drift along with the colloidal transport, instead of being sorbed onto immobile surfaces. To prevent adverse implications on the safety of the repository, the colloidal impact must be evaluated. In this paper, colloid analysis done by SKB is studied, and factors to be considered for the safety assessment of colloids are analyzed.

First, the colloid generation mechanism should be analyzed. In a cementitious repository, due to a highly alkaline environment, colloid formation from wastes may be promoted by the decomposition of organic materials, dissolution of inorganic materials, and corrosion of metals. Radiolysis is excluded when radionuclide inventory is moderate, as in the case of SKB.

Second, colloid stability should be evaluated to determine whether colloids remain in dispersion. Stable colloids acquire electric charges, allowing particles to continuously repel one another to prevent coagulation. Thus, stability depends on the pH and ionic condition of the surroundings, and colloid composition. For instance, under a highly alkaline cementitious environment, colloids tend to be negatively charged, repelling each other, but Ca^{2+} ion from cement, acting as a coagulant, makes colloid unstable, promoting sedimentation. As in the case of SKB, the colloidal impact is assumed negligible in the silo, BMA, and BTF due to their extensive cement contents, but for BLA, with relatively less cement source, the colloidal impact is a potential concern.

Third, colloid mobility should be assessed to appraise radionuclide release via colloid transport. The mobility depends on the density and size of colloids, and flow velocity to commence motion. As a part of the assessment, the filtration effect should also be included, which depends on pore size and structure. As in the case of SKB, due to static hydraulic conditions and engineering barriers, acting as efficient filters, colloidal transport is expected to be unlikely.

In the domestic underground repository, the highly alkaline environment would lead to colloid formation, but due to high Ca^{2+} concentration and low flow velocity, colloids would achieve low stability and mobility, thus colloidal impact would be a minor concern. In the future, with further detailed analysis of each factor, waste composition, and disposal condition, reliable data for safety evaluation could be generated to be used as fundamental data for planning waste acceptance criteria.

Keywords: Colloid, Radioactive waste, LILW disposal facility, Safety assessment

A Study on Plasma Torch Melter's Optimum Start-up Operation Condition

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A plasma torch is a kind of equipment that utilizes an electric arc to dissociate a gas and transfer an electric energy to the gas to generate very high temperature flame. KHNP-CRI has been developed the Plasma Torch Melter (PTM) to reduce radioactive waste disposal volumes and drop the radiation level of wastes. As you guess, there is required condition for proper start-up operation like current, voltage, plasma gas flow, cooling water flow, temperature in melter and so on. Thus, the optimum start-up operation condition of plasma torch will be estimated experimentally in this paper.

Keywords: Plasma torch, Start-up operation condition, Current, Voltage, Plasma gas flow, Cooling water flow, Temperature

Efficient Method for Separating Suspended Soil and Strontium From Aqueous Solutions Using Magnetic Flocculant

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^{90}Sr is considered a hazardous radionuclide due to its long half-life of 28.8 years, radiotoxicity, and potential to bioaccumulate in various organisms. In the environment, strontium typically exists as divalent cation Sr^{2+} or in different complexes, depending on the environmental physical and chemical factors. Despite its mobility, Sr^{2+} transport remains affected by adsorption from solid phases, such as soil and sediments. This research aimed to investigate the efficiency of a magnetic flocculant (MNP/IF) in separating suspended soil and Sr^{2+} from a soil suspension. MNP/IF was prepared via the electrostatic interaction between magnetite particles and an inorganic flocculant (IF) composed of CaCO_3 and Na_2SO_4 . Analysis of the physical properties of MNP/IF confirmed that MNP/IF was successfully imparted with magnetism and had excellent adsorption capacity for Sr^{2+} . The optimal MNP/IF dosage for the sedimentation of suspended soil was determined to be 0.3 g/g (mass ratio of flocculant to soil). The lower the pH, the more favorable the flocculation-sedimentation process of the suspended soil by MNP/IF, since Ca^{2+} and Mg^{2+} , which are the most common strong flocculators, were further eluted from IF under acidic conditions. Besides, MNP/IF exhibited outstanding removal performance for Sr^{2+} , with maximum adsorption capacities of 163.6 mg/g observed during the flocculation-sedimentation reaction of suspended soil. The adsorption of Sr^{2+} exhibited consistency with the Langmuir model and followed pseudo-second-order kinetics. These findings suggest that MNP/IF can be used for the simultaneous removal of suspended soil particles and Sr^{2+} from a radioactive soil suspension.

Acknowledgements

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Keywords: Soil suspension, Strontium, Turbidity, Flocculation, Magnetic separation

4분과

중저준위폐기물관리(Poster)

Development of CaO-based Pellet for Tc-99 Capture

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The disposal of spent nuclear fuel (SNF) poses a significant challenge due to its high radioactivity and heat generation. However, SNF contains reusable materials, such as uranium and trans-uranium, which can be recovered through aqueous reprocessing or pyrochemical processes. Prior to these processes, voloxidation is necessary to increase reaction kinetics by separating fuels from cladding and reducing the particle size. In the voloxidation, uranium dioxide (UO_2) from SNF is heated in the presence of oxygen and oxidized to triuranium octoxide (U_3O_8), resulting a release of gaseous fission products (FPs), including technetium-99 (Tc-99), which poses a risk to human health and the environment due to its high mobility and long half-life of 2.1×10^5 . To date, various methods have been developed to capture Tc in aqueous solutions. However, a means to capture the gaseous form of Tc (Tc_2O_7) is essential in the voloxidation.

Due to the radioactive properties of technetium isotopes, rhenium is often used as a substitute in laboratory settings. The chemical properties of rhenium and technetium, such as their electronic configurations, oxidation states, and atomic radii, are similar and these similarities indicates that the adsorption mechanism for rhenium can be analogous to that for technetium.

In the previous study, a disk-type adsorbent based on CaO developed was effective in capturing Re. However, this study lacked sufficient data on the chemical properties and capture performance of the adsorbent. Furthermore, the fabrication of disk-type adsorbents is time-consuming and requires multiple steps, making it impractical for mass production.

This study introduces a simple and practical method for preparing CaO-based pellets, which can be used as an adsorbent to capture Re. The results provide a better understanding of the adsorption behavior of CaO-based pellets and their potential for capturing Tc-99. To the best of our knowledge, this is the first study to apply a CaO-based pellet to capture Re and investigate its potential for capturing Tc-99.

Keywords: Spent nuclear fuel, Capture, Technetium, Rhenium, CaO-based pellet

Effect of Initial Concentration and Absorbed Dose on Decomposition of Oxalic Acid by Gamma Irradiation

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The radiolytic decomposition of oxalic acid was investigated using gamma irradiation for decontamination of nuclear power systems. The study used high-purity analytical grade chemicals, with initial concentrations of oxalic acid prepared at 1, 2, 5, and 10 mM, and the initial pH was adjusted to 2-3 at each test condition. Gamma irradiation was performed using a high-level Co-60 source, and absorbed doses were 5, 10, 20, 30, and 50 kGy. The results showed that the efficiency of gamma irradiation decreased with longer gamma exposure time, and the G-value increased with the initial concentration of oxalic acid. Interestingly, the G-value decreased with accumulated radiation dose, but the removal increased. The dose constant ranged from 0.1695 to 0.0536 kGy⁻¹ at different initial concentrations, and the G-value was inversely proportional to the dose constant. The study concluded that oxalic acid was successfully degraded by gamma irradiation, and 92% removal was obtained at the initial concentration of 10 mM. The mineralization of oxalic acid at higher concentrations was more difficult due to the great number of generated intermediates.

Acknowledgements

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Keywords: Radiolytic decomposition, Oxalic acid, Gamma irradiation, G-value

Drop Test of IP-2 Packages for Decommissioning Wastes of Nuclear Power Plant

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Transport packages have been developed to transport the decommissioning waste from the nuclear power plant. The packages are classified with Type IP-2 package. The IAEA requirements for Type IP-2 packages include that a free drop test should be performed for normal conditions of transport.

In this study, drop tests of the packages were performed to prove the structural integrity and to verify the reliability of the analysis results by comparing the test and analysis results. Half-scale models were used for the drop tests and drop position was considered as 0.3 m oblique drop on packages weighing more than 15 tons. The strain and impact acceleration data were obtained to verify the reliability of the analysis results. Before and after the drop tests, radiation shielding tests were performed to confirm that the dose rate increase was within 20% at the external surface of the package. Also, measurement of bolt torque, and visual inspection were performed to confirm the loss or dispersion of the radioactive contents.

After each drop test, slight deformations occurred in some packages. However, there was no loss of pretension in the lid bolts and the shielding thickness was not reduced for metal shields. In the package with concrete shield, the surface dose rate did not increase and there was no cracks or damage to the concrete. Therefore, the transport packages met the legal requirements (no more than a 20% increase of radiation level and no loss or dispersion of radioactive contents). Safety verifications were performed using the measured strain and acceleration data from the test, and the appropriate conservatism for the analysis results and the validity of the analysis model were confirmed. Therefore, it was found that the structural integrity of the packages was maintained under the drop test conditions. The results of this study were used as design data of the transport packages, and the packages will be used in the NPP decommissioning project in the future.

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Keywords: Drop test, Transport package, IP-2 package, Decommissioning waste, Safety verification

Fabrication of a Cement Waste Form for Final Disposal of Spent Ion Exchange Resins

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Withdrawal

A Preliminary Study on Non-Fuel Radioactive Waste Stored in Spent Fuel Pool of PWR

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The treatment of waste generated during operation as a part of preparation for decommissioning is coming to the fore as a pending issue. Non-fuel waste stored in the spent fuel pool (SFP) of PWRs in Korea includes Dummy fuel, damaged fuel rod storage container, reactor vessel specimen cask, spent in-core instrumentation, spent control element assemblies, spent neutron source assemblies, burnable poison rods, etc. In order to treat such waste, it is necessary to classify radioactive waste level and analyze kinds of nuclide in accordance with legal requirements. In order to solve the problem, the items that KHNP-CRI is trying to conduct like followings. First, KHNP-CRI will identify the current status of non-fuel waste stored in the SFP of all domestic nuclear power plants. In order to consider the treatment of non-fuel waste, it is essential to know what kind of items and how many items are stored in the SFP. Second, to identify the dimension and characteristics of non-fuel waste stored in the SFP would be conducted. The configuration of non-fuel waste is important information to handle them. Third, the way to handle non-fuel waste would be deduced including analysis of their dimension, whether the equipment should be developed to handle each kind of non-fuel waste or not, how to transport them. In order to classify radioactive waste level and analyze the nuclide for the non-fuel waste, handling tools and the cask to transport them into the facility which nuclide analysis is able to be performed would be required. Fourth, the nuclide analysis technology would be identified. Also, domestic holding technology would be identified and which technology should be developed to classify the radioactive waste level for the non-fuel waste would be deduced. This preliminary study will provide KHNP-CRI with the insight for the nuclide analysis technology and future work which is following action for the non-fuel waste. Based on the result of above preliminary study, the feasibility of the research for the treatment of non-fuel waste would be evaluated and research plan would be established. In conclusion, the treatment of non-fuel waste stored in the spent fuel pool of domestic PWR should be considered to prepare the decommissioning. KHNP-CRI will identify the quantity, the dimension and kinds of non-fuel waste in the SFP of domestic PWR. Also, the various nuclide analysis technology would be identified and the technology which should be developed would be defined through this preliminary study.

Keywords: Nuclide analysis, Non-fuel waste, Spent fuel pool, Spent control element assembly

A Comprehensive Overview for the Purification of Radioactive Wastewater Contaminated With Uranium Through Adsorption Process

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The rise of nuclear power plants to meet escalating global energy needs has made environmental pollution including the contamination of uranium due to improper disposal of radioactive wastewater during uranium milling and mining processes. Adsorption, a water purification method known for its fast kinetics, high selectivity, and ease of use, has emerged as a popular choice for the treatment of radioactive wastewater. In response to the critical need for the purification of radioactive wastewater contaminated with U(VI), this review provides a comprehensive summary of the various types of materials, synthetic methods, and adsorption mechanisms used for the purification process. The materials are categorized into four main groups: organic, inorganic, composite/nanomaterials, and framework materials. To enhance the adsorption capacity for U(VI), researchers have explored physical and chemical modifications as well as the development of organic-inorganic hybrids. The improved adsorption performance resulting from these modifications is mostly attributed to electrostatic interaction, surface complexation, and ion exchange mechanisms. However, despite the present understanding of the processes involved, further research is still needed to fully determine the optimal approach for purifying contaminated radioactive wastewater.

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Keywords: Uranium, Contamination, Adsorption, Review

Design of Cooling Water System of Vitrification Facility

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After melting glass at a high temperature of about 1,100 degrees in the Cold Crucible Induction Melter (CCIM) of the vitrification facility, radioactive waste is fed into the CCIM to vitrify radioactive waste. Accordingly, since the metal sector of the CCIM contacts the high-temperature molten glass, cooling water is supplied to continuously cool the metal sector. The cooling system is divided into primary and secondary cooling water systems. The primary cooling water flows inside the metal sector of the CCIM to maintain the metal sector within normal temperature, thereby forming a glass layer between the metal sector and the high-temperature melting glass. The secondary cooling system is a system that cools the primary cooling water that cools the metal sector, and removes heat generated from the primary cooling system. In addition, it is designed to stably supply cooling water to the secondary cooling water system through an emergency cooling water system so that cooling water can be stably supplied to the secondary cooling water system in the event of secondary cooling water loss. Therefore, it is designed to maintain the facility stably in the event of loss of cooling water for the CCIM of the vitrification facility.

Keywords: Cold crucible induction melter, Primary and secondary cooling water system

Design of Control System of Vitrification Facility

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A vitrification facility control area is formed to control and monitor the vitrification facility process, and the control system is designed to manage the vitrification facility more safely and effectively. The control system is largely composed of a process control system and an off-gas monitoring system. The process control system is operated so that operation variables can be maintained in a normal state even in normal and transient conditions, and is designed so that the vitrification facility can be stably maintained in the event of an abnormality in the facility.

The process control system consists of Programmable Logic Controller (PLC) and Local Control Panel (LCP), which controls and monitors each unit device. In addition, operation variables are provided to the operator so that the operator can manage operation variables during process control in a centralized manner for the operation of the vitrification facility. The off-gas monitoring system is operated to monitor whether the off-gas discharged to the environment is stably maintained within the standard level, and the off-gas is monitored through an independent monitoring system.

Keywords: Vitrification facility, Process control, Off-gas

A Study on the Pelletization Using Roll Compaction and Volume Reduction Comparison of Non-conformity Radioactive Wastes for Disposal

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When decommissioning and operating nuclear power plants, a lot of radioactive waste in concentrated waste powder, slurry, sludge, and powder is generated. The radioactive waste, non-conformity for disposal, cannot be treated or disposed of, but is currently being stored instead. To dispose of the waste, the waste can be solidified by mixing with an appropriate solidification agent. However, when the solidification agent and powder particles are mixed as in the conventional method, the final volume of the waste form to be disposed of increases. In order to solve this problem, in this study, volume reduction was achieved, compared to the existing powder, by applying the roll compaction technology to mold the radioactive waste into compressed pellets.

Soil, concrete, concentrate waste, and contaminated soil powder were used as test materials, and pellets were prepared under different operating conditions. Subsequently, a compressive strength test was performed to confirm the integrity and optimal process conditions of the manufactured pellets. However, in order to perform the compressive strength test, the upper and lower surfaces of the pellets must be horizontal, but the pellet has the shape of two tetrahedrons joined together. Hence, test specimens for measuring compressive strength were prepared by making a surface treatment jig. The compressive strength test showed a high strength of 5.20~28.20 MPa. The process conditions showing high compressive strengths were selected as the optimal process conditions. Finally, the volume reduction ratios were calculated by measuring the weight, density and volume of the manufactured pellets. The degrees of volume reduction of the manufactured pellets compared to the existing powder were checked. When the roll gap was 0 mm, the average reduction ratios of the test materials were 3.7 for the soil, 4.0 for the concrete, 4.6 for the concentrate waste, and 3.8 for the contaminated soil. When roll gap was 1 mm, the ratios were 2.7 for the soil, 2.9 for the concrete, 3.4 for the concentrate waste, and 2.8 for the contaminated soil. Therefore, from a conservative point of view (Roll gap = 1 mm), when powdered waste is formed into pellets, it means that the volume is reduced by 1/2.7 for soil, 1/2.9 for concrete, 1/3.4 for concentrated waste, and 1/2.8 for contaminated soil.

Acknowledgements

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Keywords: Non-conformity radioactive waste, Roll compaction, Pelletization, Volume reduction

A Study of Technology for Monitoring Volatile Substance While Treating Radioactive Wastes Using Plasma

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Radioactive waste must be stored for at least 300 years and must bear astronomical costs. In addition, unexpected potential risk factors are also a considerable burden. In the case of low-level radioactive waste, combustible and liquid low-contamination radioactive waste can be treated relatively easily through high-temperature plasma which the volume can be reduced by 1/250 and the weight by 1/30. It is possible to permanently dispose of the ash leached after plasma treatment in a more stable manner compared to the conventional methods. Types of low-level combustible radioactive waste, including paper, vinyl, clothing, filters, and resins, account for more than 30% of the total waste volume. Furthermore, high-temperature plasma treatment of low-level radioactive waste from petrochemical plants and medical institutions have many advantages, namely astronomical cost savings, securing free space in existing storage facilities, and improving the image of nuclear energy.

Korea is preparing to decommission the Kori No. 1 nuclear power plant, and small and medium-sized enterprises and related organizations are conducting various studies to incinerate radioactive waste. In foreign countries, Britain began incineration technology in the 1970s, and Plasma Energy Group, LLC, headquartered in Florida, USA, physically changed the molecular structure of the material by combining plasma chambers and plasma arcs and obtained a patent application in 1992. Germany was approved for operation in 2002, and Switzerland completed a trial run of a plasma technology-based facility in 2004. Important radionuclides in terms of radioactive gas waste treatment include inert gases, radioiodine, and radioactive suspended particles. Gas waste is compressed in a compressor through a surge tank in the gas waste treatment system and filters at each stage. After that, the short-life nuclide is naturally collapsed for 30 to 60 days in the storage or activated carbon adsorbent in the attenuation tank and released through HEPA filters. The radioactive concentration at discharge is monitored and managed using continuous monitoring equipment, and the oxygen concentration is managed in the gas waste treatment system to prevent explosion risk.

The problem of radioactive waste disposal is not only a problem for people living in the present era, but also a big social issue that brings a burden to future generations. While interest in plasma treatment is increasing from the decommissioning of the Kori Unit 1. In Korea, it is showed that there is a lack of systematic management and research especially in the radioactive volatile gases fields, that's why I propose some ideas as follows.

First, the government and related institutions should invest to the continuous radioactive monitoring system to produce and distribute continuous radioactive monitoring facilities with an affordable price. Second, it is recommended that radioactive waste incineration would be connected to the GRS system of the plant's gas radwaste treatment system, and radioactive volatile materials should be monitored through continuous monitoring system.

Third, radioactive volatile materials generated according to the temperatures and times during plasma incineration treatment are different. Therefore, prior classification of each expected radioactive volatile substance must be performed before incineration.

Keywords: Volatile material, Plasma, Radioactive Wastes, Incineration, and HEPA filters

Features of Liquid Radwaste Level Measurement Based on Differential Pressure

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Level measurement of liquid radwaste is essential for inventory management of treatment system. Among various methods, level measurement based on differential pressure has many advantages. First, it is possible to measure the liquid level of the system regardless of liquid type. Second, as the instrument doesn't need to be installed near the tank, there is no need to contact the tank when managing it. Therefore, workers' radiation dose from the system can be decreased. Finally, although it depends on the accuracy, the price of the instrument is relatively low. With these advantages, in general, liquid radwaste level in a tank is measured using differential pressure in the treatment system.

Not only the advantages described above, there are some disadvantages. As the liquid in the system is waste, it is not pure but has some suspended materials. These materials can be accumulated in tanks and pipes where the liquids move to come into direct contact with pneumatic pipes that are essential in differential pressure instruments. As a result, in case of a treatment using heat source, the accumulated materials may become sludge causing interference in pneumatic pipes. And this can change the pressure which also affects the level measured. In conclusion, in case of liquid storage tanks in which the situation cannot be checked, the proficiency of an operator becomes important.

Keywords: Differential pressure, Liquid tank, Level measurement, Radwaste treatment system

Evaluation of Applicability of Plasma Torch Melting System for Liquid Waste

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In KHNP CRI, the PTMs (plasma torch melting system) was developed as a treatment technology of a wide variety of radioactive wastes generated by nuclear power plants. The facility is made of melting zone, thermal decomposition zone, melt discharge zone, waste feeding device, MMI, and off-gas treatment system. In this study, demonstration test was conducted using NaOH solution as liquid waste to evaluation the applicability of the PTM system. For demonstration test of NaOH solution treatment, the plasma melting zone is sufficiently pre-heated by the plasma torch for 5 hours. The temperature inside the plasma melting zone is about 1,600°C. The NaOH solution as simulant was put into the thermal decomposition zone by the spray feeding device with the throughput of maximum 30 liter/hour. During the test, the power of plasma torch is about 100 kW on the transferred mode. The 160 liters of liquid waste was treated for 500 minutes. After the demonstration test, the final product in the form of salt was remained in the melting zone, and the disposal of the final product are still under consideration.

Keywords: 100 kW PTM system, Liquid waste, Demonstration test, NaOH solution

Long-term Integrity of Plasma Torch Melting Facility for Radioactive Waste

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Plasma torch melting has been considered as a promising treatment technology for radioactive waste generated by nuclear power plants. The IAEA reported in 2006, the plasma melting technology could be treated regardless of the type of radioactive wastes such as combustible, non-combustible and liquid. Also, the technology has the advantage of being an eco-friendly technology. It emits less harmful gases such as NO_x , SO_x , HCl and CO because it does not use fossil fuels.

In KHNP CRI, the plasma torch melting system was developed as the new radioactive waste treatment technology. In this study, to evaluate the long-term integrity of the new facility, a demonstration test with concrete as a simulant was carried out for about 3 days. For the 3 days, the evaluation was conducted in view of abnormal shutdown, soundness of waste feeding device, electrode consumption, and so on.

Keywords: Plasma torch melting facility, Treatment technology, Long-term integrity, Demonstration test

Development of HVAC Filter Processing Technology Using Plasma Torch Melting Technology

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Plasma melting technology is a high-temperature flame of about 1,600°C or higher generated using electrical arc phenomena such as lightning, and radioactive waste generated during the operation and dismantling of nuclear power plants is not classified according to physical characteristics. It is a technology that can meet waste disposal requirements through treatment and reduction. Plasma torch melting technology was used for volume reduction and stable treatment of HVAC filters generated from nuclear power plants HVAC (Heating Ventilation and Air Conditioning). filter was treated by placing 1 to 3 EA in a drum and injecting it into a plasma melting furnace at 1,500°C, and the facility was operated without abnormal stop. A total of 132.5 kg of filter was treated, and the high-temperature melt was normally discharged four times. It was confirmed that the plasma torch melting facility operates stably at 500 LPM of nitrogen and 370-450 A of current during filter treatment. Through this study, the possibility of plasma treatment of filters generated at nuclear power plants has been confirmed, and it is expected that stable disposal will be possible in the future.

Keywords: Plasma torch melting technology, HVAC filter, Waste, Melt

Status Update for Management of Contaminated Water in Fukushima Daiichi NPP

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Japan's government has announced plan to release the contaminated water stored from the tanks of the Fukushima Daiichi nuclear power plant site into the sea in June. The contaminated water is treated by SARRY (Cesium removal facility) and ALPS (advanced liquid processing system) to remove 62 radionuclide containing Cesium, Strontium, Iodine, and so on using filtration, precipitation (or co-precipitation) and adsorption for other nuclides (except for H-3 and C-14). The total amount of the contaminated water stored at tanks is 1,328,508 m³ (as of March 23, 2023). Currently, three ALPS systems which are existing ALPS, improved ALPS, high performance ALPS have been operated to meet the regulatory standard for release to the sea. According to the release plan, they have announced that 30 nuclides and H-3 concentration of the contaminated water will be measured and assessed before/after the discharge of the contaminated water into the sea. Before the release, the contaminated water is re-treated by reverse osmosis membrane facility and additional ALPS. And then, the water will be diluted with seawater more than 100 times. The diluted water will then move through an undersea tunnel and be released about 1 kilometer off the coast.

Keywords: SARRY, ALPS, Fukushima daiichi NPP, Contaminated water, Management of contaminated water, 30 Nuclides, H-3

Results of Improvement of Internal Monitoring Camera in Plasma Torch Melting Facility

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By developing plasma torch melting technology in 1996, our company has developed the first generation 150 kW ('96~'02), the second generation 500 kW ('08~'12), and the third generation MW plasma torch melting facility ('14~'18), and completed facility upgrading ('20~'23). The MW plasma torch melting facility is equipped with CCTV to monitor waste input, melting, torch integrity, and melt discharge. The lens is installed inside a metal housing made of stainless steel to prevent damage caused by external impacts and high temperatures, and supplies nitrogen to prevent cooling and lens contamination. As a result of the demonstration test, as the temperature inside the melting furnace increased after starting the plasma torch, the resolution decreased along with noise in the CCTV, and facility monitoring was difficult due to high temperatures and foreign substances (fume). Based on the test results, CCTV was changed to a non-insertion type that was not directly exposed to high temperatures, and a filter (quartz) was additionally applied to monitor the melt smoothly. As a result of applying the newly manufactured CCTV to the demonstration test, smooth monitoring ability was confirmed even at normal operating temperature (above 1,500°C). Through this facility improvement, the operation convenience of the plasma torch melting facility has been secured, and it is expected that it will be able to operate stably during long-term continuous operation in the future.

Keywords: Plasma torch melting technology, CCTV, Lens, Melt

Suggestion of Improvement of Slag Container for Plasma Torch Melting Facility

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Currently, KHNP-CRI has developed 100 kW plasma torch melting facility to reduce the amount of radioactive waste in nuclear power plant. Plasma torch melting technology uses electric arc phenomena like lightning to melt the target material at a high temperature of about 1,600°C. The technology is applicable to treatment for various types of waste such as combustible, non-combustible and mixed wastes. The volume reduction ratio by the technology is respectively expected to be about 1/60 of combustible wastes and about 1/5 for non-combustible wastes. It is important to discharge the melt without problems in the melting technology. In general, molten slag has properties such as high viscosity and quick solidification. Because of the properties, when discharging into slag container, the final product is accumulated like a mountain. To improve this problem, there is three suggestions; 1) rotation of the slag container, 2) vibration of the slag container, and 3) heating of the slag container.

Keywords: Plasma torch melting, Discharging, Three suggestions, Slag container

Homogeneity Evaluation of Geopolymer Waste Form Containing Radioactive Spent Ion Resins Using LIBS and Statistical Approach

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The homogeneity of radioactive spent ion exchange resins (IERs) distribution inside waste form is one of the important characteristics for acceptance of waste forms in long-term storage because heterogenous immobilization can lead to the poor structural stability of waste form. In this study, the homogeneity of metakaolin-based geopolymer waste form containing simulant IERs was evaluated using a laser-induced breakdown spectroscopy (LIBS) and statistical approach. The cation-anion mixed IERs (IRN150) were used to prepare the simulant spent IERs contaminated by non-radioactive Cs, Fe, Cr, Mn, Ni, Co, and Sr (0.44, 8.03, 6.22, 4.21, 4.66, 0.48, and 0.90 mg/g-dried IER, respectively). The K₂SiO₃ solution to metakaolin ratio was kept constant at 1.2 and spent IERs loading was 5wt%. For the synthesis of homogeneous geopolymer waste form, spent IERs were mixed with K₂SiO₃ solution and metakaolin first, and then the fresh mixture slurry was poured into plastic molds (diameter: 2.9 cm and height: 6.0 cm). The heterogeneous geopolymer waste form was also fabricated by stacking two kinds of mixtures (8wt% IERs loading in bottom and 2wt% in top) in one mold. Geopolymers were cured for 7d (1d at room temperature and 6d at 60°C). The hardened geopolymers were cut into top, middle, and bottom parts. The LIBS spectra and intensities for Cs were obtained from the top and bottom of each part. Cs was selected for target nuclide because of its good sensitivity for measurement. Shapiro-Wilk test was performed to determine the normality of LIBS data, and it revealed that data from the homogeneous sample is normal distribution (p-value = 0.9246, if p-value is higher than 0.05, it is considered as normal distribution). However, data from the heterogeneous sample showed abnormal distribution (p-value = 7.765×10^{-8}). The coefficient of variation (CoV) was also calculated to examine the dispersion of data. It was 31.3% and 51.8% from homogeneous and heterogeneous samples, respectively. These results suggest that LIBS analysis and statistical approaches can be used to evaluate the homogeneity of waste forms for the acceptance criterion in repositories.

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Keywords: Homogeneity of solid waste form, Spent ion exchange resin, Geopolymer, Normality test, LIBS

Considerations for Tracking and Management of Radioactive Waste From the Decommissioning of Nuclear Power Plants

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Korea currently has two permanent shutdown Nuclear Power Plants (NPPs), and the decommissioning project is expected to begin soon, starting with the first commercial NPP. The decommissioning project will eventually be the disposal of radioactive waste in the final stage of the work, and in that respect, proper tracking and history management should be well established in the management of waste. This is in line with the guidelines that regulatory agencies should also properly manage radioactive waste. Therefore, this study intends to examine the factors that should be considered in terms of tracking and management of radioactive waste in decommissioning nuclear facilities.

The starting and final point of tracking radioactive waste generated during decommissioning is the physical inventory of the current as-is state and the final container. In this respect, the tracking of waste starts from the beginning of the dismantling operation. Thus, at the stage of approval of the decommissioning work, it may begin with an ID scheme, such as the functional location in operation for the target System, Structure, and Components (SSCs). As the dismantling work progresses, SSCs will be classified by nature and radiological level, which will be placed in containers in small packaging units. At this time, the small package should be given an ID. After that, the dismantling work leads to the treatment of waste, which involves a series of operations such as cutting, decomposition, melting, and decontamination. Each step in which these tasks are performed will be placed in a container, and ID assignment is also required. Until now, the small packaging container is for transfer after each treatment, and it is placed in the storage container in the final stage, at which time the storage container also gives a unique ID.

Considerations for follow-up management were reviewed assuming solid waste, which is the majority of dismantled radioactive waste considered in this study. The ID system should be prepared from the start of the dismantling work, ID generation of the small transporting container and ID generation of the final disposal container during the intermediate waste treatment process, and each ID generation of the previous stage should be linked to each generation stage. In addition, each ID must be generated, and the definition of the grant scheme and attributes is required.

Keywords: Decommissioning, Waste tracking, Disposal, Waste classification

Review of the Applicability of Low-Level Waste Disposition Options During the Decommissioning of Nuclear Facilities

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In nuclear decommissioning projects, past and present projects in the world, an important area to be managed is waste management. The management of waste should be done with various aspects of consideration in mind from the moment it occurs from the cutting and dismantling of Systems, Structures, and Components (SSCs). Therefore, this study aims to discuss the disposition considerations for the efficient management of low and very low level waste that is expected to be generated in large quantities and to examine its applicability to domestic nuclear facilities.

As for waste management, radioactive wastes begin to be generated when SSCs are dismantled, so waste management should be carried out as a result of dismantling activities. In addition, the waste is stored in the final disposal container and transferred to the storage or disposable facilities. In order to store in the final disposal container or transfer container, it will have to be classified by radioactive level. From the perspective of waste classification, wastes below the low level can be divided into low levels, very low levels, and clearance in Korea. Therefore, as an important point of waste management, when SSCs are dismantled, the work process must be carried out until the final disposal in accordance with the disposition strategy based on the waste classification.

As a disposition strategy, the process presented by the IAEA can be referred. The materials to be dismantled for the first time are largely divided into radioactive and suspected radioactive materials. After going through the dismantling process, three criteria are considered to satisfy the disposition option: unconditional release criterion, conditional criterion, and radioactive waste. The types of waste below the final low level are classified into two types as unconditional, two types as conditional, and low and very low levels.

In this study, six disposition options are reviewed, including unconditioned reuse and disposal, conditional reuse and disposal, and disposal of VLLW and VLW. Options for radioactive waste may be subject to operational criteria and may need to be supplemented in terms of the acceptance criteria in the repository. In the case of the conditional option, the clearance criterion can be applied, but considering the decommissioning characteristics, it is an option that can be used for nuclear industry, and specific reuse scenarios should be supplemented through discussions with the regulatory agency. In addition, it seems that the unconditional option needs to establish a corresponding criterion.

Keywords: Decommissioning, Waste management, Disposition option, Waste classification

Development of Radioactive Waste Solidification Process Technology and Analysis of Acceptance Criteria for Solids at Domestic and Foreign Disposal Facilities

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The acceptance criteria for low and intermediate level radioactive waste disposal facilities in Korea to regulate that homogeneous waste, such as concentrated waste and spent resin, should be solidified. In addition, solidification requirements such as compressive strength and leaching test must be satisfied for the solidified radioactive waste solidified sample. It is necessary to develop technologies such as the development of a solidification process for radioactive waste to be solidified and the characteristics of a solidification support. Radioactive waste solidification methods include cement solidification, geopolymer solidification, and vitrification. In general, low-temperature solidification methods such as cement solidification and geopolymer solidification have the advantage of being inexpensive and having simple process equipment. As a high-temperature solidification method, there is typically a vitrification. Glass solidification is generally widely used as a stabilization method for liquid high-level waste, and when applied to low- and intermediate-level radioactive waste, the volume reduction effect due to melting of combustible waste can be obtained.

In this study, the advantages and disadvantages of the solidification process technology for radioactive waste and the criteria for accepting the solidified material from domestic and foreign disposal facilities were analyzed.

Keywords: Solidification, Concentrated waste, Spent resin

A Study on Evaluation Methods of Secondary Waste Generated From Decommissioning of Large Concrete Drums

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In the Kori-1 radioactive waste storage, the concentrated waste and spent resin drums generated in the past are repacked and stored in large concrete drums. In order to dispose of radioactive waste generated before the establishment of the waste acceptance criteria, it is necessary to develop a large concrete drum treatment and waste treatment process to evaluate disposal suitability and secure technology that meets the latest technical standards. In addition, for worker safety and waste reduction, it is important to develop secondary waste treatment technology generated during waste treatment.

In this study, the types and characteristics of secondary wastes that can be generated when large concrete drums are decommissioned were investigated. In addition, considering the characteristics of possible secondary wastes, suitable treatment methods and characteristic evaluations were analyzed. We plan to develop an optimal process for secondary waste treatment in consideration of on-site work space, economic feasibility, and safety.

Keywords: Large concrete drum, Secondary waste, Concentrated waste, Spent resin

Review of the Environmental Risk in the Thermal Treatment Process of Dry Active Waste

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Thermal treatment, such as combustion, is the most effective way to solve the spatial problem of radioactive waste disposal. Existing incineration technology has the problem of discharging harmful pollutants (CO₂ and dioxin, etc.) into the environment. Therefore, it was evaluated the validity of the thermal treatment process that can reduce the volume of dry active waste (DAW) in an eco-friendly. In addition, the stability of the alternative incineration process under development was evaluated by evaluating the emission of harmful pollutants to the environment during the thermal treatment process.

We selected 14 samples identical to those discarded by each nuclear power plant (Kori, Saeul, Wolsong, Hanbit, Hanul). And EA (Elemental Analysis) analysis was performed on each sample. As a result, excluded samples containing wastes containing POPs (Persistent Organic Pollutants) such as PCBs (Polychlorinated Biphenyls), which could generate harmful pollutants during thermal treatment, and halogenated organic wastes such as PVC (Polyvinyl Chloride). In addition, the thermal treatment conditions for the four DAWs were derived by Thermogravimetric Analysis/Differential Thermal Analysis (TG/DTA) analysis. At this time, Py-GC/MS analysis was performed at the temperature at which each waste causes thermal decomposition (cotton is 437°C, paper is 562°C, latex glove is 430°C, plastic bag is 485°C). As a result of analyzing the exhaust gas produced during thermal decomposition, about 77.0% of the cotton was Benzoic acid series, the paper was 41.1% Glucopyranose series, and 15.8% hydroxy acetaldehyde. Latex glove was identified to be 45.9% and 19.2% for Limonene and 2-methyl-1, 3-Butadiene, and for plastic bags, Octacosanol and 2-octyl-1-Dodecanol were 38.8% and 15.2%. In addition, it was confirmed that dioxin and harmful heavy metals, which are discussed as environmental risks, were not detected in all samples.

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Keywords: Dry active waste, Alternative incineration, Thermal decomposition, Harmful pollutants, Exhaust gas

The Establishment of Dispersive Radioactive Waste Solidification Devices and Preliminary Performance Evaluation

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A lot of solid wastes are generated when nuclear power plant is dismantled, and a lot of treatment costs and optimal waste treatment technologies are required to treat the generated solid wastes. Currently, there is no optimized reduction and solidification technology for each characteristics of radioactive dismantling waste, so the customized treatment technology for each waste is required to respond actively to this issue. This paper shows the evaluation results of molding and sintering characteristics using preliminary sample to derive operational characteristics and improvements for powder mixing device, molding device, and sintering device manufactured for solidification of dispersible radioactive waste.

Zeolite was used as a preliminary sample for performing basic operation characteristics evaluation of each unit device. First of all, the basic operation characteristics of the powder mixing device was evaluated by analyzing the sample distribution, mixing degree, and tap density. It was confirmed that the preliminary sample was well mixed in all areas of the cylinder where the mixing was performed. In the tap density analysis, the increase effect of the volume reduction of the sample was confirmed according to the increase of the RPM speed (up to 2000 RPM). Since the particle size of zeolite sample is very small (nanometer size), the particular consistency of the change of average particle size with RPM speed couldn't be confirmed, but the uniform of particle size distribution was confirmed with RPM speed size. The basic operation characteristics of the molding device was evaluated for each mold size (ID30, ID50, ID100) according to the moisture content (0-20%) and the molding pressure condition (25-200 MPa) for the preliminary sample. In the characteristics evaluation of the sintered body, the strength of the sintered body was much higher than that of the molded body. However, it was confirmed that as moisture evaporated during the sintering process according to the moisture content contained in the molded body, the swelling occurred in the sintered body due to vapor pressure, and this caused cracks in the longitudinal or transverse direction inside and outside the sintered body. Therefore, optimal moisture content conditions for sintering should be derived.

In conclusion, if the operation characteristics and improvements of powder mixing, molding and sintering devices derived from this study are reflected and improved, it is judged that it is possible to derive the optimal process for solidification of dispersive radioactive wastes.

Acknowledgements

This work was supported by National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (RS-2022-00155257).

Keywords: Dispersible radioactive waste, Glass composite material (GCM), Disposal conditions, Solidification device

Effect of Polycarboxylic-ester (PCE) Cement Admixture on Radionuclide Release in Repository Environments

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In underground repository environments, various types of engineered barriers are installed to hinder the mobility of radionuclides. Cement admixtures, especially used to improve workability for concrete, are composed of fairly high organic molecules and have a dispersing effect through bonding with the C-S-H of the concrete. Previous studies have shown that complex-forming organics like EDTA, NTA, and ISA have a significant effect on the mobility of radionuclides, but the studies on the behavior and stability of combined complexes in hydrated cement are lacking. So, we selected a commonly used polycarboxylic-ester (PCE) type cement admixture and stable Co as a surrogate of Co-60 to perform desorption experiments from hydrated cement containing the admixture. Radioactive Co is known to be a common contaminant in nuclear fission and medical facilities and considered to exist as a relatively stable phase in repositories. In addition, the evaluation of cobalt can be a standard of safety issue for other radionuclides with the presence of cement admixture in repository. In this study, cement samples were prepared at water/cement ratio of 0.55 and cured for 28 days at 23-25°C and at least 80% of humidity with varying cement admixtures of 0.0, 0.1, and 2.0wt%. To evaluate the stability of cobalt in the weathered cement, a 0.001 M HCl solution was used to simulate cement weathering conditions on a hot plate at 60°C for 1 day using a solid/liquid ratio of 1:100. Degree of weathering was confirmed using XRD analysis. The adsorption experiments were performed by adding 0.0042 mmol of cobalt (CoCl_2 , Sigma-Aldrich, anhydrous $\geq 98.0\%$) to the weathered cement for 3 days using a platform shaker at 200 rpm, and the supernatant was separated using a syringe filter ($<0.20 \mu\text{m}$) before ICP-MS analysis to determine the amount of Co adsorption. Cobalt desorption was tested for the Co-adsorbed cement using 0.019 mmol of calcium ($\text{Ca}(\text{NO}_3)_2 \cdot 4\text{H}_2\text{O}$, Sigma-Aldrich, 99%) for 3 hours to 14 days. The results showed that adsorbed cobalt with and without cement admixture was stably bound to cement, and did not increase any noticeable Co release by 2.0wt% PCE admixture. However, additional experiments using varying contents of PCE and other admixtures should be conducted to provide a standard for assessing the safety of cement admixtures in repositories.

Acknowledgements

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Keywords: Cobalt, Polycarboxylic-ester based (PCE) copolymer, Desorption, Radionuclide, Mobility

Establishment of Infrastructure for Analysis of Characteristics of Intermediate-level Radioactive Decommissioning Waste

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In Korea, many characteristic component facilities and technologies in general experimental areas for non-radiative materials are owned by industry-academia research. Still, no characteristic analysis test technology has been developed for large, intermediate-level decommissioning waste emitted by neutron irradiation. Since Korea plans to decommission nuclear power plants in 2027, securing analysis technology for intermediate-level decommissioning waste is essential. Accordingly, the Korea Research Institute of Decommissioning (KRID) plans to secure an infrastructure (hot cell) to analyze the characteristics of intermediate-level dismantled waste. Afterward, we intend to stably dispose of the waste generated while decommissioning the current Gori Unit 1/Wolseong Unit 1 using the intermediate-level dedicated hot cell. It aims to secure high-dose/high-radiation decommissioning waste handling technology through intermediate-level hot cells for the first time in Korea, supports domestic nuclear-decommissioning projects, and secure and validate procedures related to material characteristics and nuclide analysis of intermediate-level waste. Furthermore, research on intermediate-level radioactive materials is expected to be carried out in cooperation with schools and research institutes.

Keywords: Intermediate-level radioactive material, Decommissioning waste, Hot cell, KRID

Proposal on the Cellulose Degradation Model for the Domestic 1st Phase Underground Repository

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With the recent concern regarding cellulose enhancing radionuclide mobility upon its degradation to ISA, disposal of cellulosic wastes is being held off until the disposal safety is vindicated. Thus, a rational assessment should be conducted, applying an appropriate cellulose degradation model considering the disposal environment and cellulose degradation mechanisms. In this paper cellulose degradation mechanisms and the disposal environment are studied to propose the best-suitable cellulose degradation model for the domestic 1st phase repository.

For the cellulose to readily degrade, the pH should be greater than 12.5. As in the case of SKB, 1BLA is excluded from the safety assessment because the pH of 1BLA remains below 12.5. Furthermore, despite cellulose degradation occurring, it does not always produce ISA. At low Ca^{2+} concentration, the ISA yield rate is around 25%, but at high Ca^{2+} concentration, the ISA yield rate increases up to 90%. Thus, for the cellulose to be a major concern, both pH and Ca^{2+} concentration conditions must be satisfied.

To satisfy both conditions, the cement hydration must be in 2nd phase, when the porewater pH remains around 12.5 and a significant amount of Ca^{2+} ion is leaching out from the cement. However, according to the safety evaluation and domestic research, 2nd phase of cement hydration for silo concrete would achieve a pH of around 12.4, dissatisfying cellulose degradation condition like in 1BLA. Thus, cellulose degradation would be unlikely to occur in the domestic 1st phase repository.

To derive waste acceptance criteria, a quantitative evaluation should be conducted, conservatively assuming cellulose is degraded. To conduct a safety evaluation, an appropriate degradation model should be applied to determine the degradation rate of cellulose. According to overseas research, despite the mid-chain scission being yet to be seen in the experiments, the degradation model considering mid-chain scission is applied, resulting in an almost 100% degradation rate. The model is selected because the repositories are backfilled with cement, achieving a pH greater than 13, so extensive degradation is reasonably conservative.

However, under the domestic disposal condition, where cellulose degradation is unlikely to occur, applying such model would be excessively conservative. Thus, the peeling and stopping model derived by Van Loon and Haas, which suggests 10~25% degradation rate, is reasonably conservative. Based on this model, cellulose would not be a major concern in the domestic 1st phase repository. In the future, this study could be used as fundamental data for planning waste acceptance criteria.

Keywords: Cellulose degradation, ISA, Disposal environment, Safety assessment

Development of Automated Liquid Radioactive Waste Pre-treatment Device at KAERI

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The nuclear facilities at Korea Atomic Energy Research Institute (KAERI) have generated a variety of liquid radioactive waste and most of them have low-level radioactive or lower levels. Some of the liquid radioactive waste generated in KAERI is transported to Radioactive Waste Treatment Facility (RWTF) in 20 L container.

Liquid radioactive waste transported in a 20 L container is stored in a Sewer Tank after passing through a solid-liquid separation filter. It is then transferred to a very low-level liquid radioactive waste Tank after removing impurities such as sludge through a pre-treatment device. The previous pre-treatment process involved an underwater pump and a cartridge filter device passively, but this presented challenges such as the inconvenience of having to install the underwater pump each time, radiation exposure for workers due to frequent replacement of the cartridge filter, and the generation of large amounts of radioactive waste from the filter. To address these challenges and improve efficiency and safety in radiation work, an automated liquid radioactive waste pre-treatment device was developed.

The automated liquid radioactive waste pre-treatment device is a pressure filtration system that utilizes multiple overlapping filter plates and pump pressure to effectively remove impurities such as sludge from liquid radioactive waste. With just the push of a button, the device automatically supplies and processes the waste, reducing radiation hazards and ensuring worker safety. Its modular and mobile design allows for flexible utilization in various locations, enabling efficient pre-treatment of liquid radioactive waste.

To evaluate the performance of the newly constructed automated liquid radioactive waste treatment device, samples were taken before and after treatment for 1 hour cycling and analyzed for turbidity. The results showed that the turbidity after treatment was more than about four times lower than before treatment, confirming the excellent performance of the device. Also, it is expected that the treatment efficiency will improve further as the treatment time and number of cycles increase.

Acknowledgements

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Keywords: Liquid radioactive waste, Pre-treatment, Automated device, Turbidity

A Prototype Safety Assessment Tool for a Radioactive Waste Treatment Complex Based on IAEA SADRWMS Methodology Using Microsoft Excel and Visual Basic Add-in

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Kori Unit 1 was permanently shut down in 2017 and is preparing to be dismantled. Decommissioning nuclear power plants is expected to generate a lot of decommissioning waste. Therefore, a radioactive waste treatment complex will be built on the site to safely and effectively the process of decommissioning waste generated from the Kori Unit 1, and the details are specified in the decommissioning plan. Therefore, a safety assessment should be conducted according to the facility's normal and abnormal operations to construct a radioactive waste treatment complex.

Currently, a safety assessment for a radioactive waste treatment complex can be conducted by the Safety Assessment Framework (SAFRAN) Tool based on the Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) methodology developed by the International Atomic Energy Agency (IAEA). The SAFRAN Tool can be calculated radiation dose and hazard quotient (HQ) for workers and the public under normal and abnormal conditions of the radioactive waste treatment complex. When evaluating the radiation dose for the public due to releasing radioactive materials into the air or discharging radioactive materials into liquids, the radiation dose is calculated using the amount discharged or released from the treatment complex, and the Pathway Dose Factors (PDFs) derived from the generic environmental model given in the IAEA Safety Reports Series No.19. PDFs, which reflect the specific site data rather than the generic environmental model data, should be calculated and evaluated when performing the safety evaluation of the radioactive waste treatment complex to be built on the Kori site. In addition, in the SAFRAN tool, there is an inconvenience in that it must be calculated separately by radionuclides to calculate the contribution of dose or HQ for each radionuclide.

Therefore, in this study, a safety assessment tool for a radioactive waste treatment complex was developed using Visual Basic by supplementing the limitations of the SAFRAN tool. This tool was developed to allow users to choose whether to apply PDFs based on the IAEA SRS-19 based on the generic environmental model or PDFs calculated to reflect the specific site data. Furthermore, the tool considered all types of decommissioning wastes that may occur during the decommissioning of the Kori Unit 1 and the treatment process scheduled to be introduced. Therefore, this study is expected to be used as basic data when conducting the safety assessment of radioactive waste treatment complex scheduled to be introduced in Korea.

Acknowledgements

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Keywords: Radioactive waste treatment complex, SADRWMS, SAFRAN, Pathway dose factors, Decommissioning, Safety assessment

A Strategy for Clearance Level Radioactive Waste; Used Laboratory Fume Hood

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It is important to make a strategy for clearance-level radioactive waste. Sampling and disposal plans should be drawn up with characteristics of target waste.

In this paper, a target clearance-level radioactive waste is used in a laboratory for experiments with Cs-137 and Co-60, unsealed radioactive sources with gamma radiation isotopes. Therefore, it is enough to analyze with HPGe to check the contaminant level.

The laboratory fume hood combined multiple materials, which means some are volume contamination and others are surface contamination. The wood, plastic, and drywall boards, which are absorbent volume contaminated parts and make up PVC pipes, base cabinet doors, backside baffles, etc., will be sampled with coring methods. The metals and glasses, which are unabsorbent, surface-contaminated parts, are sampled with smear methods.

The work surface, baffles, exhaust plenum, and glass sash inside parts have a high possibility of being contaminated. The hood body, flame, base cabinet, PVC pipe (the rare end of the filter), and blower transition case have a low possibility of becoming contaminated. When we checked with HPGe, except for the work surface (which was below clearance level), other parts were less than MDA.

The highest radionuclide concentration was in PVC pipe: Cs-137C 3.91E-02 (Bq/g), Co-60 4.54E-03 (Bq/g). It is less than clearance level. Therefore, the waste was applied for the clearance level radioactive wastes and got permission from the regulatory body.

Keywords: Clearance level radioactive waste, Used laboratory fume hood, Complexed material, Volume contaminated, Surface contaminated

Grouping Measures for Optimizing the Packaging and Compression Process of Spent Filters

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The spent filters used to purify radioactive materials and remove impurities from primary systems at nuclear power plants (NPPs) have been stored for long periods in filter storage rooms at NPPs due to concerns about the unproven safety of the treatment method, absence of disposal facilities, and risk of high radiation exposure. In the storage room at Kori Unit 1, there are approximately 227 spent filters of 9 different types. The radiation dose rates of filters range from 0.01 to 500 mSv/hr.

Recently, a comprehensive plan has been established for the treatment and disposal of radioactive waste that has not yet been treated to facilitate decommissioning of NPPs. As a follow-up measure, compression and packaging optimization processes are being developed to treat the spent filters. KHNP plans to dispose of the spent filters after compressing, packaging, and immobilizing them. However, the spent filters are currently stored without being sorted by type or radiation intensity. If the removal and packing of the filters are done randomly without a plan for the order of withdrawal and subsequent processes, issues may arise such as a decrease in drum loading efficiency and exceeding the dose limit of the package.

In this study, the number of drums needed to pack the spent filters was calculated, considering the filter size, weight, quantity, dose rate, shielding thickness of drum, and loadable quantity in a shielding drum (SD). Then, the spent filters that can be loaded on each drum were classified into one group. In addition, the withdrawal order for each group was set so that the filter withdrawal, compression, and packaging processes could be performed efficiently. The spent filter groups are as follows: (1) compression/12 cm SD (17 groups), (2) compression/16 cm SD (6 groups), (3) non-compression/intermediate storage container (17 groups, additional radiation attenuation required due to high dose rate), and (4) unclassified (5 groups, determined after measurement due to lack of filter information). The withdrawal order of the groups was determined based on several factors, including visual identification of the filter, ease of distribution after withdrawal, work convenience, and safety.

Due to the decay of radioactivity over time, the current dose rate of the spent filters is expected to be much lower than at the time of waste generation. Therefore, in the future, sample filters will be taken from the storage room to measure their radioactivity and radiation dose rate. Based on these measurements, a database of radiological characteristics for the 227 filters will be created and used to revise the filter grouping.

Keywords: Radioactive waste, Spent filters, Packaging, Compression, Grouping

A Study on the Characteristics of Reverse Polarity Plasma Torch According to Inter-electrode Conditions

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DC plasma torch is reported as a technology that can be treated regardless of waste types because it can select transfer or non-transfer operation modes depending on the electrical conductivity of waste. Thanks to this characteristic, countries that operate nuclear power plants such as Switzerland, Japan, and Taiwan have developed high-power DC plasma torch to dispose of radioactive waste. And also in Korea, a plasma torch is being developed to dispose of radioactive waste. This study was conducted to investigate the characteristics of the reverse polarity plasma torch according to the conditions of inter-electrodes. The inter-electrodes of plasma torch used in the study was designed to be 25, 37 mm in diameter and 180 to 400 mm in length. As a result of the test, it was confirmed that the smaller the diameter and the longer the length of the inter-electrode, the more advantageous it was to achieve a high output power. And it was confirmed that the power torch would be 500 kW when the diameter of the inter-electrode was 25 mm, the length was 400 mm, and the current was 500 A.

Keywords: Plasma, Non-transfer torch, Melting, Inter-electrode, Radioactive waste

A Study on the Plasma Torch Output According to Operation Mode and Conditions

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It is important that the plasma torch used in the waste treatment field has a high output to increase throughput. In order to increase the output of the plasma torch, there is a method of increasing the current or extending the length of the plasma arc. Among these methods, high power can be easily achieved simply by increasing current, but it is difficult to ensure electrode life. Therefore, it is necessary to check the appropriate current and arc length conditions to achieve high power and stable operation. In this paper, the power performance according to the arc length, current, and operation mode was confirmed in the transfer mode plasma torch. The test conditions are the distance (arc length) between the plasma torch and the external electrode was set to 5-180 mm, and the current was set to be in the range of 90-460 A. As a result of the test, it was confirmed that the reverse polarity operation had a maximum output of 159 kW depending on the arc length and current, and the positive polarity operation had a maximum output of 138 kW. Through this result, it was confirmed that the arc length had an effect on increasing the output, and that the reverse polarity operation had a longer arc than the positive polarity operation.

Keywords: Plasma, Transferred type torch, Melting, Reverse polarity, Positive polarity, Radioactive waste

A Study on the Reverse Polarity Plasma Torch for the Control of Cathode Point

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Depending on the type of waste, DC plasma torch uses a transfer type operation for conductive waste and a non-transfer type operation for non-conductive waste. The transfer mode plasma torch can secure high throughput because the arc directly contacts the object and has high thermal efficiency. However, since the non-transfer mode does not have a higher thermal efficiency than the transfer mode, higher output is required to secure high throughput. A method of increasing the output of the plasma torch is increasing the current or extending the length of the plasma arc. However, the method of increasing the current affects the life of the electrode, and there is a limit to extending the arc length in the positive polarity plasma torch. Therefore, it is effective to design the plasma torch with reverse polarity to secure life and extend the arc length. In the reverse polarity plasma torch, the front electrode serves as the cathode, and the cathode point is not easy to control compared to the anode point, which may cause abnormal arcing and damage the plasma torch. This paper was conducted to investigate the conditions for securing the safety of these non-transferable reverse polarity plasma torch. The plasma torch is designed to have an output of 100 kW or less and to use the detachable nozzle to control the cathode point. The test showed that the shape of the nozzle prevented the cathode point moving outside of plasma torch and the excessive extension of the arc. Thanks to this, it was confirmed that plasma could be stably formed and abnormal arcing could also be prevented.

Keywords: Plasma, Non-transfer torch, Reverse polarity, Cathode point, Radioactive waste

Prediction of Radioactive Inventory Using the DTC Method for Long-term Storage Spent Filters

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Commercial operation of KORI Unit 1 ended in 2017, and the final decommissioning plan is currently under approval from the KINS. In order for the dismantling waste to go to the repository, it is judged that the radioactive waste generated during the commercial operation should be treated and disposed in advance. Among these radioactive wastes, spent filters contain various radionuclides. The radiation dose rate from the radiation coming out of the filters ranges from a low dose rate to high dose rate. Therefore, in order to handle the spent filters, a remote processing system is required to reduce the radiation exposure of workers.

This paper evaluates the radioactive inventory of filters that are stored in the filter room at the KORI unit #1. For this purpose, a method for predicting the radioactivity of each nuclide in the filter, based on the radiation dose rate, has been described using the MicroShield code, which is a commercial shielding code. The information on the filters in the field has only the creation date, type, size, and surface dose rate. In order to evaluate the radioactivity inventory using such limited data, it is possible to know the nuclide radioactivity ratio in the filter. We took out some of the filters stored on site and measured from using the ISCOS system, a gamma nuclide analyzer. The radioactivity of each nuclide in the filter was inferred by modeling with the MicroShield code, based on the radiation dose rate and the radioactivity value of each nuclide measured in the field.

Keywords: ISOCS, MicroShield, Spent Filter

A Conceptual Study on Radioactive Waste Feeding System Using Plasma Torch Melter

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KHNP-CRI has developed small-capacity and Mega-Watt Class PTM (Plasma Torch Melter) for the purpose of reducing the volume of radioactive waste and immobilizing or solidifying radioactive materials. About 1 MW PTM is a treatment technology that operates a plasma torch and puts drum-shaped waste into a melter and radioactive waste in the form of slag is discharged into a waste container. The small-capacity PTM is a treatment technology that operates a plasma torch and puts small amounts of radioactive waste by directly putting it into the melter through a waste input machine. Mega-Watt Class PTM was able to inject radioactive waste in drums, so it was disposed of without backlogging. On the other hand, The small-capacity PTM put radioactive waste without a package, and the waste input was blocked. If even small-capacity PTM put radioactive waste in the form of small packages such as drums, it is expected that various types of radioactive waste can be processed for a long time. Packaging also reduces the risk of radioactive contamination.

Keywords: Plasma torch melter (PTM), Radioactive waste disposal, Radioactive waste feeding system

Verification of Integrated Decontamination Equipment Automation Performance to Remove Concrete Surface Contamination

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Concrete radioactive waste is divided into surface-contaminated concrete and activated concrete, and although the generation rate varies depending on the operating conditions of the nuclear power plant, it is reported that the amount of surface-contaminated concrete generated is greater. It is reported in the 'US-NRC Inventory Report' that 99% of radionuclides in surface-contaminated concrete are distributed within 1 mm of the surface. Since concrete radioactive waste accounts for a large amount of generation after metal radioactive waste, it is necessary to reduce the amount of radioactive waste disposal by applying appropriate treatment techniques to surface-contaminated concrete.

In this study, a similar contamination environment work space with the size of 5.4 (W) × 3.6 (L) × 2.5 (H) [m] in which concrete specimens can be fixed on the wall and floor was established. And an integrated decontamination equipment was verified the automation performance for 'location accuracy', 'radioactive contamination level measurement' and 'concrete surface laser scabbling'.

It was confirmed that the average was 8.3 [mm] in the evaluation of the 'location accuracy' for the remote control and movement of the integrated decontamination equipment.

For performance verification of 'radioactive contamination level measurement' and 'laser scabbling', it were used that size of 30×30×8 [cm] ordinary concrete specimens and concrete radioactively contaminated with Co-60 below the regulatory exemption concentration.

'Radioactive contamination level measurement' is measured as much as the set range, divided and display the measured values in different colors on the map of the control program. Ordinary concrete specimens are 0.066~0.089 μ Sv/hr, and contaminated concrete specimens are 0.107~0.121 μ Sv/hr, and it was confirmed that they are expressed in different colors on the map.

For 'laser scabbling', the performance according to the laser scabbling speed and reproducibility with ordinary concrete specimens was verified. As a result, a weight change of up to 1.48 kg was confirmed. Contaminated concrete specimens were subjected to a direct method using a surface contamination detector and an indirect method using a smear paper to measure surface contamination before and after scabbling, and the debris generated after scabbling was analyzed using HPGe.

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Keywords: Decommissioning, Concrete surface decontamination, Radioactive waste reduction, Remote control, Robot

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Development of Calibration Drum Production Method for Non-Destructive Assay System for Packaging Spent Filters

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In order to permanently dispose of radioactive waste drums generated from nuclear power plants, disposal suitability must be demonstrated and the nuclides and radioactivity contained in the waste drums, including those in the shielding drums, must be identified. At present, reliable measurements of the nuclide concentration are performed using drum nuclide analysis devices at power plants and disposal facilities during acceptance inspection. The essential functions required to perform nuclide analysis using the non-destructive assay system are the correction for self-attenuation and the dead time correction. Until now, measurements have mainly been performed for drums containing solid waste such as DAW drums using SGS calibration drums with ordinary iron drums. However, for drums containing non-uniform radioactive waste, such as waste filters embedded in cement within shielding drums, a separate calibration drum needs to be produced.

In order to produce calibration drums for shielded and embedded waste drums, the design considered the placement of calibration sources, setting of shielding thickness, correction for medium density, and cement mixing ratio. Based on these considerations, three calibration drums were produced.

First, a shielding drum with an empty interior was produced. Second, a density correction drum filled with cement was produced to create apparent density on the surface of the shielding drum. Third, a physical model drum was produced containing a mock waste filter and cement filled in the shielding drum.

Keywords: Spent filter, SGS, Calibration

A Study on Representative Sampling Plan for Spent Filters in Radioactive Waste Drums

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Spent filters contained in drums of radioactive waste generated from nuclear power plants are contaminated with various radioactive isotopes due to their use in various water purification processes in the system. Radiation doses from the spent filters can vary from low to high levels. To dispose of drums containing spent filters as radioactive waste, the inventory of radioactive isotopes in the filters must be determined. Two methods for determining the inventory are indirect measurement using scaling factors and direct analysis of filter samples.

This study suggests a method to determine the appropriate sample size for each drum based on the number of filters stored in the drum, when direct analysis is used to determine the inventory of radioactive isotopes. In particular, Visual Sample Plan (PNNL) software's Item Sampling function was used to calculate the sample size, considering the confidence level and minimum acceptable coverage rate.

As a result, assuming that the number of filters packed per drum ranges from a minimum of 1 to a maximum of 30, the study suggests that a full inspection is required for drums containing 9 or fewer filters, while drums containing 10 filters should be sampled with 9 samples, 11 filters with 10 samples, 12-13 filters with 11 samples, 14-16 filters with 12 samples, 19-22 filters with 14 samples, 23-26 filters with 15 samples, and 27-30 filters with 16 samples.

Keywords: Spent filter, Sampling, Visual sample plan

Sensitivity Analysis for Reactive Transport Model in Concrete Disposal System

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LILW disposal repository in Gyeongju, South Korea is considered with a concrete mixture that uses Ordinary Portland Cement (OPC) partially substituted with supplementary cementitious materials (SCMs). The degradation of cementitious materials that result from chemical and physical attacks is a major concern in the safety of radioactive waste disposal. We present a reactive transport model utilized as one of the geochemical simulation approaches for the timescales of concern that range from hundreds to thousands of years.

The purpose of this study is to investigate the sensitivity of parameters in concrete disposal systems and to evaluate the influence of various assumptions on the chemical degradation of the systems using a reactive transport model. A reactive transport model in the concrete disposal vault was developed to evaluate the behavior of engineered barriers composed of cementitious materials. The sensitivity analysis was performed using reactive transport models through the coupling between COMSOL and PHREEQC. The databases selected for the analysis are the Thermochemie database presented by ANDRA.

Among many variables considered, two variables that can highly affect chemical degradation were selected for detailed sensitivity analysis for dealing with uncertainties. This is important because the chemical degradation mechanism is generally sensitive to precipitation and diffusion coefficient. The first factor is precipitation, which might be the most important factor in chemical degradation because it acts as a calcium leaching of cementitious materials in a disposal system in a highly alkaline environment, increasing the porosity of the system. To predict the change in annual precipitation, the measurement of the precipitation observatory station in the nearest area of Gyeongju for the past 80 years was collected. The second factor is the diffusion coefficient, which plays an essential role in the durability of the concrete disposal system, promoting the decalcification of cementitious minerals, accelerating system degradation, and increasing the porosity of its system, thereby facilitating the migration of radionuclides. The diffusion coefficient values used in studies similar to this work were calculated and evaluated using the box-and-whisker method. The results of the sensitivity analyses for the reactive transport model in the concrete disposal system will be presented. The sensitivity cases show that the results obtained are much more sensitive to changes in transport parameters.

Keywords: Reactive transport model, Concrete disposal system, Chemical degradation, Cementitious materials

A Study on the Treatment Method and Packaging Concept by of Radioactive Waste

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Decommissioning of Nuclear Power Plant (NPP) projects in South Korea starts with permanent shutdown of Kori unit 1 and Wolsung unit 1. It is important to establish a treatment and disposal method for radioactive waste generated during the decommissioning of the nuclear power plants. Large quantities of the wastes during decommissioning of NPP are generated in a short period of time and the wastes have various types and characteristics. For efficient decommissioning of NPP process, the radioactive waste is classified by types and each treatment method and packaging concept is presented respectively in this paper.

Radioactive waste generated during decommissioning of NPP is classified into reactor vessel, reactor internals, metals, Dry Active Waste (DAW), concrete, spent fuel storage rack, spent resin and spent filter, etc., and the packaging concept for each type should be established to meet the waste acceptance criteria. Major waste acceptance criteria requirements include nuclides concentration, filling rate, free water, surface radiation dose rate and weight.

Radioactive waste containers can be classified into packaging containers, transport containers, and disposal containers. The packaging container is used to contain, transport, and store radioactive waste within the radiation control area, and a control number has been assigned as a radioactive waste drum after the final treatment has been completed. The transport container is used for transporting radioactive waste filled-containers from a radiation control area through an uncontrolled area.

In this paper, the concept of disposal of dismantled radioactive waste and packaging methods were reviewed in comprehensive consideration of domestic radioactive waste transport and storage regulations, permanent disposal environment, and development status of large containers.

Keywords: Decommissioning radwaste, Packaging, Disposal process, Waste container

Review on Overseas Vitrification Technology Commercialization Methodology

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KHNP's vitrification technology introduced a commercialized vitrification facility to the Hanul nuclear power site after a commercialization test through a lab test and a pilot plant at KHNP-CRI. France's ANADEC (consortium with CEA, Orano, ECM Technologies and Andra) conducted a feasibility evaluation from FY2018 to FY2021 to apply In-Can vitrification, which was developed to treat Fukushima Effluent Treatment Waste (FETW) such as carbonate slurry and ferric slurry generated from ALPS (Advanced Liquid Processing System-Multi Radionuclides Removal) facilities for waste treatment in Fukushima, Japan.

For commercialization, the following method was used. First, through the Laboratory scale studies, the possibility of high waste loading (60wt% in dry mass) of slurry on borosilicate matrix was tested. In addition, the volatility of radionuclide was evaluated through radionuclides surrogates with a Bench-scale mockup and glass discharge (100 kg) was evaluated through In-Can vitrification process verification. The feeding system was improved through a pilot scale test, and finally, glass discharge (300 kg) was evaluated after large amount of waste was treated through an industrial prototype (Full-scale) at the CEA Marcoule site (France).

Keywords: Vitrification, In-Can, Commercialization

A Study on the Optimization of the Package of Reactor Internal Vessels of Nuclear Power Plants Using Demplus

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As the decommissioning of nuclear power plants progresses, interest in the inevitably generated radioactive waste is also increasing. Especially, because the containers of ILW packages are significantly more expensive than the containers of LLW packages, the special attention should be focused on minimizing the number of the containers of ILW packages. The radiation dose limit for packaging of ILW shall not exceed 2 mSv/h and 0.1 mSv/h on contact and at 2 m, respectively in South Korea. Meanwhile, The DEMplus provides various environmental geometry and all properties such as materials, absorptions, and reflections and the estimation of the radiation dose rates is based on the radiation interactions of the designed 3D geometry model. With the consideration of the radiation dose rate by using DEMplus and its strategy of packaging and cutting plan, the number of containers for ILW packages generated from decommissioning of Reactor Vessel Internal (RVI) of a nuclear power plant that has been in operation for decades was optimized in this paper. The modular shielded containers (MSC) with shielding inserted were used for radioactive wastes that require shielded packaging. In order to verify the accuracy of the estimated radiation dose rate by using DEMplus, the estimated results were compared with those obtained using MicroShield. The trends of the estimated radiation dose rates using DEMplus and the estimation of MicroShield were similar to each other. The results of this study demonstrated the feasibility of using DEMplus as a means of estimating the radiation dose limit in packaging plan of the radioactive waste.

Keywords: Intermediate level waste (ILW), Radioactive waste package optimization, Reactor vessel internal (RVI), Modular shielded containers (MSC), DEMplus, MicroShield

A Case Study on Plasma Torch Melting Facility's Cost Estimation in Terms of Radioactive Waste Disposal

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A disposal of radioactive wastes is one of the critical issues in our society. Considering upcoming plans for dismantling of nuclear power plants, this problem is inevitable and should be discussed very carefully. There are variety of methods to handle with radioactive wastes, including Incineration, conventional gasification and plasma gasification. Among them, plasma gasification process is in the limelight due to its eco-friendly & stable operation, and large volume reduction effects. However, a fatal disadvantage is that it consumes more electric power than other methods, this leaves us a question of whether this process is indeed economical. Within the scope of this paper, I would like to introduce 4 cases which plasma facilities were evaluated economically in worldwide, and reach the conclusion on the economic feasibility of plasma process.

Keywords: Disposal of radioactive wastes, Incineration, Conventional gasification, Plasma gasification, Cost estimation, Economic feasibility

ISO 24389 Series and Its Utilization for Radioactive Waste Data Analysis Framework

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Withdrawal

A Study on Effect of Taking out the Non-Fuel Assembly (NFA) in SFP & NFA Treatment, Storage and Disposal Method

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Most of the spent nuclear fuel generated by domestic nuclear power plants (NPPs) is temporarily stored in wet storage which is spent fuel pool (SFP) at each site. Currently, in case of Kori Unit 2, about 93.6% of spent nuclear fuel is stored in SFP. Without clear disposal policy determined for spent nuclear fuel, the storage capacity in each nuclear power plant is expected to reach saturation within 2030.

Currently, the SFP stores not only spent fuel but also various non-fuel assembly (NFA). NFA apply to all device and structures except for fuel rods inserted in nuclear fuel assembly. The representative NFA is control element driving mechanism (CEDM), in-core instrument (ICI), burnable poison, and neutral resources.

Although these components are irradiated in the reactor, they do not emit high-temperature heat and high radiation like nuclear fuel, so if they are classified as intermediate level waste (ILW) and low level waste (LLW) and moved outside the SFP, positive effects such as securing spent fuel storage space and delaying saturation points can be obtained.

Therefore, this study analyzes the status of spent fuel and Non Fuel Assembly (NFA) storage in SFP of domestic nuclear power plants. In addition, this study predict the amount of spent fuel and NFA that occur in the future. For example, this study predicts the percentage of current and future ICIs and control rods in the SFP when stored in the spent fuel storage rack. In addition, the positive effects of moving NFA outside the SFP is analyzed.

In addition, NFA withdrawn from SFP is classified as ILW & LLW according to the classification criteria, and the treatment, storage, and disposal methods of NFA will be considered. The study on the treatment, storage, and disposal methods of NFA is planned to be conducted by applying the existing KN-12 & KN-18 containers and ILW & LLW containers being developed for decommissioning waste.

Acknowledgements

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Keywords: Spent nuclear fuel (SNF), Non-fuel assembly (NFA), SFP capacity, NFA disposal, NFA treatment, NFA storage

A Study on the Operation of Scaling Factors for Radioactive Waste During Permanent Shutdown Prior to Decommissioning of Nuclear Power Plants

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Domestic nuclear power plants developed the scaling factors for the radioactive waste generated from 2004 to 2008 for the first time. Afterwards, the effectiveness of continuous application of the scaling factors have been evaluated for newly generated radioactive waste over the past two years. It was confirmed that most of the initially developed scale factors were effective within a factor of 10 principle, which is an acceptable range. The scaling factors were updated using the analysis data base from 2004 to 2016. A scaling factor refers to the calculated abundance ratio between Key (Easy-to-Measure) and DTM (Difficult-to-Measure) nuclide at the time of generation of radioactive waste based on the source term in the reactor of an operating nuclear power plant. The effectiveness of continuous application of scaling factors can be evaluated at regular intervals regardless of operation status or when there are events that change scaling factors during nuclear power plant operation, such as zinc injection, large-scale facility replacement, and long-term shutdown etc. Even in the case of a permanently shut down nuclear power plant in which fuel is withdrawn from the reactor and generation of new nuclides by nuclear fission and radiation has stopped, periodic verification is conducted to confirm whether the scaling factor developed before permanent shutdown can be effectively applied to the radioactive waste generated after permanent shutdown. However, depending on the nuclear power plant decommissioning strategy or conditions, the period of permanent shutdown prior to decommissioning can be very long, so preparations are needed to ensure the appropriateness of scaling factor operation. In the case of domestic nuclear power plants, Kori Unit 1, a light water reactor, was permanently shut down in June 2017, and as a heavy water reactor nuclear power plant, the permanent shutdown of Wolseong Unit 1 was finally decided in December 2019 after twists and turns including large-scale facility replacement and long-term shutdown. In this paper, we have predicted when the scaling factors will change significantly due to radioactive decay and the difference in half-life between the Key and DTM nuclides over time after permanent shutdown. We also have tried to find appropriate countermeasures for the operation of scaling factors during permanent shutdown period, such as updating scaling factors or applying correction factors.

Keywords : Permanent shutdown, Scaling factor, Radioactive decay, Corrected scaling factor

Review on the Prerequisites for Disposal Delivery of Polymer Concrete High Integrity Container (PC-HIC) Packages

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Licensing for the application of the Polymer Concrete High Integrity Container (PC-HIC) to nuclear power plants has been completed or is in progress. Approval for the expanded application to all domestic nuclear power plants has been completed to utilize the 860 L PC-HICs for the 2nd stage surface repository, and the regulatory body is reviewing the license application to use the 510 L PC-HICs for the 1st stage underground repository in the representative nuclear power plants. The 860 L PC-HICs, which have been licensed for all domestic nuclear power plants, will be used for safe storage management and disposal of low-dose dried concentrate waste and spent resin, and a total of 100 units is expected to be supplied to representative nuclear power plants that have been licensed first. The 510 L PC-HICs are planned to be used for underground disposal of high-dose spent resin and dried concentrate waste. Prior to the application of PC-HICs to nuclear power plants and disposal to the repository, it is necessary to establish realistic and reasonable requirements through close consultations between waste generator and disposal operators to ensure the suitability for disposal of PC-HIC packages and to carry out disposal delivery and acceptance work. Since the Polymer Concrete High Integrity Container (PC-HIC) has long-term integrity of more than 300 years and the barrier does not temporarily collapse, spent resin and dried concentrate waste, which are radioactive wastes to be solidified, can be disposed of much more safely in PC-HIC packages than solidified types. Acceptance criteria for the PC-HIC packages should be prepared fully reflecting the advantages of PC-HIC, and quality assurance methods for physical/chemical/radiological characterization results based on the Waste Certification Program (WCP) should be supported. In addition, infrastructure should be secured for safe transportation, handling, and storage of the PC-HIC packages. In this paper, we have tried to find a reasonable acceptance criteria, quality assurance method, and infrastructure level according to the dose and disposal conditions of PC-HIC packages.

Keywords: PC-HIC package, Waste certification plan (WCP), Acceptance criteria, Quality assurance

Development of Characterization Database Management System for Radioactive Waste Disposal

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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. Approximately 5,000 tons of waste will be generated when the contaminated buildings are demolished. Various types of radioactive waste are generated in large quantities during the operation and decommissioning of nuclear facilities, and in order to dispose of them in a disposal facility, it is necessary to physico-chemically characterize the radioactive waste. The need to transparently and clearly conduct and manage radioactive waste characterization methods and results in accordance with relevant laws, regulations, acceptance standards is emerging. For radioactive waste characterization information, all information must be provided to the disposal facility by measuring and testing the physical, chemical, and radiological characteristics and inputting related documents. At this time, field workers have the inconvenience of performing computerized work after manually inputting radioactive waste characterization information, and there is always a possibility that human errors may occur during manual input. Furthermore, when disposing of radioactive waste, the production of the documents necessary for disposal is also done manually, resulting in the aforementioned human error and very low production efficiency of numerous documents. In addition, as quality control is applied to the entire process from generation to treatment and disposal of radioactive waste, it is necessary to physically protect data and investigate data quality in order to manage the history information of radioactive waste produced in computerized work. In this study, we develop a system that can directly compute the radioactive waste characterization information at the field site where the test and measurement are performed, protect the stored radioactive waste characterization data, and provide a system that can secure reliability.

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Keywords: Waste management, Waste disposal, Database system, Waste traceability, Waste characterization

Commercial Operation Status on LAW Vitrification in US DOE Hanford Site

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Hanford site has been operated since 1943 to produce the plutonium for nuclear weapons. Significant amount of radioactive wastes was generated by the nuclear weapons production process. The radioactive wastes are stored in 177 aged underground tanks. Due to the risk of leakage into the air and the Columbia River, the US DOE and EPA, and Washington State Department of Ecology organized the Tri-Party Agreement (TPA) to clean-up the Hanford site in 1989.

The LAW (low-activity waste) vitrification facility named WTP (Waste Treatment Plant) is plan to vitrify about 212 million liters of radioactive waste. The US DOE announced that the world's largest melter to vitrify the LAW was heated up on October 8, 2022.

Keywords: LAW vitrification, Hanford site, Commercial operation, Radioactive waste

Comparison of Oversea Melting Technology in Fukushima Radioactive Waste Management

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After the Fukushima accident, significant amount of radioactively contaminated waste has been generated with 50~250 m³/day and stored in tanks of the Fukushima Daiichi nuclear power plant site. The contaminated water is treated by various treatment facility such as KURION, SARRY, Reverse Osmosis, and ALPS to remove 62 radioactive nuclides except H-3. For the contaminated water treatment process, massive secondary wastes such as sludge, spent adsorbent, and so on as by-product are being generated by the facilities. In Japan, to treat the secondary wastes, melting technologies such as GeoMelt, In-can vitrification and Cold Crucible Induction Melting vitrification are considered as a candidate technologies.

In this study, the technologies were reviewed, and the advantage and disadvantage of each technology were evaluated as the candidate technologies for treatment of the secondary wastes.

Keywords: Melting technology, Vitrification, Fukushima, Contaminated water, Secondary wastes

Treatment of Radioactive Spent Filter Using Plasma Torch Melting Technology

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Air conditioning facilities in nuclear power plants use pre-filters, HEPA filters, activated carbon filters, and bag filters to remove radionuclides and other harmful substances in the atmosphere. Spent filters generate more than 100 drums per year per a nuclear power plant and are stored in temporary radioactive waste storage.

Plasma torch melting technology is a method that can dramatically reduce volume by burning and melting combustible, non-flammable, and mixed wastes using plasma jet heat sources of 1,600°C or higher and arc Joule heat using electric energy, which is clean energy.

KHNP CRI & KPS are developing and improving waste treatment technology using MW-class plasma torch melting facilities to stably treat and reduce the volume of radioactive waste. This study aims to develop an operation process to reduce the volume of bag filter waste generated from the air conditioning system of nuclear power plants using plasma torch melting technology, and to stably treat and dispose of it. It is expected to secure stability and reduce treatment costs of regularly generated filter waste treatment, and contribute to the export of radioactive waste treatment technology by upgrading plasma torch melting technology in the future.

Keywords: Radioactive spent filter, Plasma torch melting, Treatment technology, MW-class plasma torch

Development of Process Control Program for Solidification of Radioactive Sludges

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Withdrawal

~~A Review of Cellulose Degradation Environment and Conditions~~

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Withdrawal

Direct Melting of Non-combustible Wastes Using Transferred Type Plasma Torch

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In this work, we report test results for direct melting of non-combustible wastes by using a 100 kW class transferred type plasma torch. For this purpose, non-combustible wastes consisting of metals and sands were prepared, weighed and melted by a transferred arc in a ceramic crucible with inner diameter of 150 mm. Test results reveal that 75wt% M6 iron bolts mixed with 25wt% sands were completely melted down within 140 seconds at the plasma power level of 83.8 kW, producing melting speed of 100 kg/hr and volume reduction rate of 62.8%. In addition, for simulated wastes consisting of 77.3wt% metal chips and 22.7wt% sands, the volume reduction rate high than 88% was achieved at 50 kW plasma power. These results indicate that non-combustible wastes can be treated efficiently when directly melting them by using transferred type plasma torch.

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Keywords: Volume reduction, Plasma melting, Transferred arc, Non-combustible wastes

Review of Characterization Method for Final Disposal of KAERI's Radioactive Waste

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As the importance of radioactive waste management has emerged, quality assurance management of radioactive waste has been legally mandated and the Korea Radioactive Waste Agency (KORAD) established the “Waste Acceptance Criteria for the 1st Phase Disposal Facility of the Wolsong Low- and Intermediate-Level Waste Disposal Center (WAC)”, the detailed guideline for radioactive waste acceptance. Accordingly, the Korea Atomic Energy Research Institute (KAERI) introduced a radioactive waste quality assurance management system and developed detailed procedures for performing the waste packaging and characterization methods suggested in the WAC. In this study, we reviewed the radioactive waste characterization method established by the KAERI to meet the WAC presented by the KORAD.

In the WAC, the characterization items for the disposal of radioactive waste were divided into six major categories (general requirements, solidification and immobilization requirements, radiological, physical, chemical, and biological requirements), and each subcategories are shown in detail under the major classification. In order to satisfy the characterization criteria for each detailed item, KAERI divided the procedure into a characterization item performed during the packaging process of radioactive waste, a separate test item, and a characterization item performed after the packaging was completed. Based on the KAERI's radioactive waste packaging procedure, the procedure for characterization of the above items is summarized as follows.

First, during the radioactive waste packaging process, the characterization corresponding to the general requirements (waste type) is performed, such as checking the classification status of the contents and checking whether there are substances unsuitable for disposal, etc. Also, characterization corresponding to the physical requirements is performed by checking the void fraction in waste package and visual confirmation of particulate matter, substances containing free water, etc. In addition, chemical and biological requirements can be characterized by visually confirming that no hazardous chemicals (explosive, flammable, gaseous substances, perishables, infectious substances, etc.) are included during the packaging process, and by taking pictures at each packaging steps.

Items for characterization using separate test samples include radiological, physical, and chemical requirements. The detailed items include identification of radionuclide and radioactivity concentration, particulate matter identification test, free water and chelate content measurement tests, etc.

Characterization items performing after the packaging is completed include general requirements such as measuring the weight and height of packages and radiological requirements such as measurements of surface dose rate and contamination, etc.

All of the above procedures are proceduralized and managed in the radioactive waste quality assurance procedure, and a report including the characterization results is prepared and submitted when requesting acceptance of radioactive waste.

The characterization of KAERI's radioactive waste has been systematically established and progressed under the quality assurance system. In the future, we plan to supplement various items that require further improvement, and through this, we can expect to improve the reliability of radioactive waste management and activate the final disposal of KAERI's radioactive waste.

Keywords: Radioactive waste disposal, Radioactive waste acceptance criteria, Characterization items of radioactive waste, Radioactive waste quality assurance management system

Analysis of Heat Flux Distribution on Anode Melting Pool by Transferred Type Arc Plasma Torch

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Nowadays, transferred type arc plasma torches have been widely present in industrial applications, in particular, using melting pool of electrically conducting materials such as arc furnace, welding and volume reduction of radioactive wastes. In these applications, the melting pools are normally employed as an anode, thus, heat flux distributions on anode melting pool need to be characterized for optimum design of melting pool system. For this purpose, we revisited the one-dimensional model of the anode boundary layer of arcs and solved governing equations numerically by using Runge-Kutta method. In addition, the direct melting process of non-combustible wastes in the crucibles were discussed with the calculation results.

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Keywords: Volume reduction, Plasma melting, Heat flux distribution, Transferred arc

Verification Plan for Compliance With Permissible Standards for Clearance of Specified Waste

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The types of waste generated in radiation controlled areas of nuclear facilities are very diverse. Among them, the waste containing hazardous materials such as electrical equipment and fire safety equipment that do not directly handle radioactive materials is also primarily classified as radioactive waste because it was used and stored in the radiation controlled area. Such wastes include periodic consumables such as fluorescent lamps, fire extinguishers, batteries, and gas containers after use. The waste is ambiguous and cannot be easily treated as radioactive waste or waste subject to clearance, and has been stored in a radiation controlled area for a long time, and the amount is continuously increasing. The storage space is saturated and has difficulty in management. IAEA ISO-7503-2016 clearly states that surface contamination measurement can be applied to surface contamination substrates (impermeable, non-activated) instead of volume contamination measurement. In order to solve these concerns, some facilities within the Korea Atomic Energy Research Institute were selected to explore self-disposal methods based on surface contamination in consideration of the characteristics of waste and facility contamination. The surface contamination degree and qualitative gamma spectroscopic analysis were carried out by the method. First, we examined the characteristics of the facility, the history of the air pollution level of the usage/storage space, and periodic inspection records. Second, we measured the physical properties (area/weight) of the waste in the same treatment way as the existing waste. Third, gamma dose rate and surface contamination (direct/indirect method) are measured for the entire area to confirm contamination is possible. It was confirmed that the concentration standard was satisfied. In order to clarify the presence of contamination, a qualitative method of gamma nuclide analysis was also performed. All surveys/measurements of 4 types of waste at 7 facilities were performed and it was confirmed that all waste satisfies the permissible concentration standard for clearance which conservatively set at 0.1 Bq/g as the permissible concentration standard. In the future, We hope that you will use this as a reference to search for easier disposal methods for regulatory bodies and specified waste disposal methods, and contribute to reducing the amount of radioactive waste generated.

Keywords: Specified waste, Hazardous material, Surface contamination

A Convertible Plasma Torch System for Efficient Melting of Non-Metallic Waste

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Normally, non-metallic wastes, such as sands, concrete and asbestos are regarded as electrically non-conductive materials. However, when the temperatures are increased up to the melting point, their electrical conductivities can be greatly improved, flowing arc current. Accordingly, these non-metallic wastes can be efficiently treated by heating them up to the electrically conducting temperatures by using a non-transferred type plasma torch, and then, melting them completely with arc currents in transferred mode of plasma torch. For this purpose, we propose a convertible plasma torch consisting of three cylindrical electrodes (rear electrode, front electrode and exit nozzle). Compared with conventional plasma torch with two cylindrical electrodes (rear electrode and front electrode), the proposed plasma torch can provide more stable plasma jet in high powered and non-transferred mode due to the presence of exit nozzle, resulting in rapid heating of the non-conductive materials.

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Keywords: Volume reduction, Plasma melting, Convertible plasma torch, Non-conductive waste

Development of a Spatial Resolution Target for X-ray Computed Tomography Metrology: Towards Quantitative Non-Destructive Evaluation of Wasteforms

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Metakaolin-based geopolymers have shown promise as suitable candidates for ¹⁴C immobilization and final disposal. It has been shown that the physicochemical properties of metakaolin wasteforms meet, and often far exceeding, the strict compression strength and leaching acceptance criteria of the South Korea radioactive waste disposal site. However, it is not possible to analyze and characterize the internal structure of the geopolymer wasteform by conventional characterization techniques such as microscopy without destruction of the wasteform; an impractical solution for inspecting wasteforms destined for final disposal. Internal inspection is important for ensuring wastes are homogeneously mixed throughout the wasteform and that the wasteform itself does not pose any significant defects that may have formed either during formulation and curing or as a result of testing prior to final disposal.

X-ray Computed Tomography (XCT) enables Non-Destructive Evaluation (NDE) of objects, such as final wasteforms, allowing for both their internal and external, characterization without destruction. However, for accurate quantification of an object's dimensions the spatial resolution (length and volume measures) must be known to a high degree of precision and accuracy. This often requires extensive knowledge of the equipment being used, its precise set-up, maintenance and calibration, as well as expert operation to yield the best results. A spatial resolution target consists of manufactured defects of uniform dimensions and geometries which can be measured to a high degree of accuracy. Implementing the use of a spatial resolution target, the dimensions of which are known and certified independently, would allow for rapid dimensional calibration of XCT systems for the purpose of object metrology. However, for a spatial resolution target to be practical it should be made of the same material as the intended specimen, or at least exhibit comparable X-ray attenuation.

In this study, attempts have been made to manufacture spatial resolution targets using geopolymer, silica glass, and alumina rods, as well as 3D printed materials with varying degrees of success. The metakaolin was activated by an alkaline activator KOH to form a geopolymer paste that was moulded into a cylinder (Diameter approx. 25 mm). The solidified geopolymer cylinder as well as both the silica glass rod and alumina rod (Diameter approx. 25 mm) were cut to approximately 4 mm ± 0.5 mm height with additional end caps cut measuring 17.5 mm ± 2.5 mm height. All parts were then polished to a high finish and visually inspected for their suitability as spatial resolution targets.

Acknowledgements

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Keywords: X-ray computed tomography, Spatial target, Spatial resolution, Metrology, Non-destructive evaluation, Wasteform

5분과

제염해체(Oral)

Volatilization of Na and B During Vitrification of HLW and Associated Effects on Crystallization

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During nuclear waste vitrification, loss of sodium (Na) and boron (B) occurs, as these elements are highly volatile at high temperatures, which causes fluctuations in composition and consequently affects the properties of the glass products. In this study, we investigated the volatilization behaviors of Na and B from a simulated high-level waste glass as functions of heating temperature and dwelling duration. Based on the data obtained regarding the composition of Na and B and the structure of the glass, a hypothetical model was proposed to explain the volatilization behaviors of Na and B from a structural viewpoint. As the loss of Na and B during vitrification, the crystallization of the glass occurred. Thus, the crystallization behavior of the simulated waste glass upon composition deviation was studied.

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Keywords: Nuclear waste glass, Sodium, Boron, Volatilization, Crystallization

Enhancing Cs Removal From Clay Minerals Through Bead-type Adsorbent Assisted Acid Washing

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The removal of cesium (Cs) from contaminated clay minerals is still a challenge due to the limited efficiency of the process. Thus, this study aimed to enhance the removal for Cs^+ ions during the conventional acid washing process by incorporating a bead-type adsorbent. Polyacrylonitrile-based nickel potassium hexacyanoferrate (NiFC-PAN) was utilized as the Cs adsorbent to selectively adsorb Cs^+ ions in a strongly acidic solution that contained competing ions. To enable easy separation of clay particles and protect the adsorbent from harsh environmental conditions, PAN was deliberately constructed as large beads. The synthesized adsorbent (NiFC/PAN in a 2:1 ratio) displayed high selectivity for Cs^+ ions and had a maximum capacity of 162.78 mg/g for Cs^+ adsorption in 0.5 M HNO_3 solution. Since NiFC-PAN exhibited greater Cs selectivity than the clay mineral (hydrobiotite, HBT), adding NiFC-PAN during the acid washing substantially increased Cs desorption (73.3%) by preventing the re-adsorption for Cs^+ ions on the HBT. The acid treatment in the presence of NiFC-PAN also significantly decreased the radioactivity of ^{137}Cs -HBT from 209 to 27 Bq/g, resulting in a desorption efficiency of 87.1%. Therefore, these findings suggest that the proposed technique is a potentially useful and effective method for decontaminating radioactive clay.

Acknowledgements

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Keywords: Cesium, Clay mineral, Desorption, Adsorbent, Separation efficiency

A Study on the Characteristic Methods of Laser Cutting Secondary Products

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Laser cutting technology capable of remote cutting is being developed to reduce radiation exposure to workers and minimize secondary waste generation when dismantling highly polluted nuclear power plant facilities (reactors, pressurizers, steam generators, coolant pumps, etc.). Laser cutting proceeds in air or water, and at this time, secondary products containing radioactive materials are inevitably generated. In air cutting, dust and aerosol are generated, and in underwater cutting, aerosol, water vapor, dispersed particles (colloid, suspension), sediment (dross, sediment), and radioactive waste liquid are generated. Dispersed particles float in the form of fine particles in water, increasing the turbidity of water as cutting progresses, hindering work, and aerosols contain micrometer-sized particles together with water vapor, which can threaten the safety of workers. Particles dispersed in water and aerosol are within 10% of the mass ratio among secondary products, but the volume they occupy is very large, which can have a significant impact on the environment as well as a burden on treatment capacity. Various characterization methods are being developed to diagnose the generation mechanism and physical and chemical properties of laser cutting secondary products in real time and to secure technologies for collecting and removing dispersed particles and aerosols in water.

This study introduces a real-time laser cutting secondary product characteristic evaluation method that can identify the key mechanisms of secondary product generation by analyzing the plasma formation process on laser cutting surface and behavior of aerosol, underwater dispersed particles produced by secondary products, as well as physical and chemical properties in real time with various measurement technologies such as Optical Emission Spectrometer (OES), Particle Size Analyzer (PSA), Scanning Electron Microscopy (SEM), X-ray Diffraction (XRD), Energy-dispersive X-ray spectroscopy (EDX), Transmission electron microscopy (TEM) and Inductively Coupled Plasma Time-of-Flight Mass Spectrometry (ICP-TOF-MS).

Keywords: Laser, Cutting, Characterization, Aerosol, Colloid

Radionuclide and Heavy Metal Extraction Using Functionalised Magnetic Nanoparticles for Environmental Remediation

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The ability to both assay the presence of, and to selectively remove ions in a solution is an important tool for waste water treatment in many industrial sectors, especially the nuclear industry. Nuclear waste streams contain high concentrations of heavy metals ions and radionuclides, which are extremely toxic and harmful to the environment, wildlife and humans. For the UK nuclear industry alone, it is estimated that there will be 4.9 million metric tonnes of radioactive waste by 2125, which contains a significant number of toxic radionuclides and heavy metals. This is exacerbated further by increased international growth of nuclear new build and decommissioning. Efforts to remove radionuclides have been focused on the development and optimisation of current separation and sequestering techniques as well as new technologies. Due to the large volumes of waste the techniques must be economical, simple to use and highly efficient in application.

Magnetic nanoparticles (MNPs) offer a powerful enhancement of normal ion exchange materials in that they can be navigated to specific places using external magnetic fields and hence can be used to investigate challenges such as, pipework in preparation of decommissioning projects. They also have the potential to be fine-tuned to extract a variety of other radionuclides and toxic heavy metals. It has been demonstrated that with the right functional groups these particles become very strongly selective to radionuclides, such as Uranium. However, this new technology also has the potential to effectively aid nuclear waste remediation at a low cost for the separation of both radionuclides and heavy metals.

In this work, we investigate the origin of the selectivity of superparamagnetic iron oxide nanoparticles (SPIONs) to Uranium by making systematic changes to the existing surface chemistry and determining how these changes influence the selectivity. Identifying the mechanism by which selected common nuclear related metals, such as Na(I), K(I), Cs(I), Ca(II), Cu(II), Co(II), Ni(II), Cd(II), Mg(II), Sr(II), Pb(II), Al(III), Mn(II), Eu(III) and Fe(III), are sorbed will allow for specific NP-target (nanoparticle) ion interactions to be revealed. Ultimately this understanding will provide guidance in the design of new targeted NP-ligand constructs for other environmental systems.

Keywords: Environmental remediation, Decontamination, Nanomaterials, Radionuclides, Heavy metals

Development of RUCAS: Verification of Dose Assessment Model for Recycling Scenarios With Shielding Compared With RESRAD-RECYCLE and MicroShield®

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RUCAS (Recycling-Underlying Computational Dose Assessment System), a dose assessment program based on the RESRAD-RECYCLE framework, is designed to evaluate dose for recycling scenarios of radioactive waste in metals and concrete. To confirm the validity of the recycling scenarios provided by RUCAS, comparative evaluations will be conducted with RESRAD-RECYCLE for metal radioactive waste recycling scenarios and with MicroShield® for concrete radioactive waste recycling scenarios.

In the evaluation of metal recycling scenarios without shielding, RUCAS showed similar results when compared to both MicroShield® and RESRAD-RECYCLE. This validates the function of dose assessments using RUCAS for metal recycling scenarios. However, when shielding was present, RUCAS produced results that were comparable to MicroShield®, but differed from those of RESRAD-RECYCLE. The underestimation of dose values up to $1.66\text{E}+08$ times difference by RESRAD-RECYCLE could potentially decrease reliability and safety in evaluated doses, further emphasizing the importance of RUCAS. Because validation is also necessary for the expanded calculation capabilities resulting from methodological changes of RUCAS (i.e., various radiation source geometries), based on prior validations, it was determined that additional validations are required for different radiation source materials and shielding conditions.

In case where the radiation source and shielding materials were identical, RUCAS and MicroShield® produced similar results according to both the Kalos et al. (1974) and Lin and Jiang (1996) methodologies. This demonstrates that the differences in methodology are inconsequential when considering the same source and shielding materials. However, when the atomic number of the radiation source materials was larger than that of shielding material (HZ-LZ condition), RUCAS obtained results similar to MicroShield® only for the Kalos et al. (1974) methodology. While Lin and Jiang (1996) methodology yield higher results than MicroShield®. Lastly, in case where the atomic number of the radiation source material was smaller than that of the shielding material (LZ-HZ condition,) both methodologies yielded results comparable to MicroShield®.

In conclusion, the validity of RUCAS's shielding calculations has been verified, confirming improvements in dose assessment compared to RESRAD-RECYCLE. Additionally, we observed that shielding effectiveness calculations differ depending on the methodology of build-up effect. If the validity of these methodologies is confirmed, it is expected that selecting the most advantageous methodology for each condition will enable more rational dose assessments. Consequently, in future research, we plan to evaluate the validity of Lin and Jiang (1996) methodology using particle transport codes based on the Monte Carlo method, such as MCNP and Geant 4, rather than MicroShield®.

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Keywords: Clearance, Recycling, Dose assessment, Shielding, RESRAD-RECYCLE

Real-time Analysis of Metal Cutting Generated Aerosol Diameter Using Plasma Arc Torch

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During the decommissioning of a nuclear power plant, the structures must be dismantled to a disposal size. Thermal cutting methods are used to reduce metal structures to a disposal size. When metal is cut using thermal cutting methods, aerosols of 1 μm or less are generated. To protect workers from aerosols in the work environment during cutting, it is necessary to understand the characteristics of the aerosols generated during the cutting process. In this study, changes in aerosol characteristics in the working environment were observed during metal thermal cutting.

The cutting was done using the plasma arc cutting method. To simulate the aerosols generated during metal cutting in the decommissioning of a nuclear power plant, a non-radioactive stainless steel plate with a thickness of 20 mm was cut. The cutting condition was set to plasma current: 80 A cutting speed: 100 mm/min. The aerosols generated during cutting were measured using a high-resolution aerosol measurement device called HR-ELPI+ (Dekati®). The HR-ELPI+ is an instrument that can measure the range of aerodynamic diameter from 0.006 μm to 10 μm divided into 500 channels. Using the HR-ELPI+, the number concentration of aerosols generated during the cutting process was measured in real-time.

We measured the aerosols generated during cutting at regular intervals from the beginning of cutting. The analyzed aerosol concentration increased almost 10 times, from 5.22×10^6 [$1/\text{cm}^3$] at the start of cutting to 6.03×10^7 [$1/\text{cm}^3$] at the end. To investigate the characteristics of the distribution, we calculated the Count Median Aerodynamic Diameter (CMAD), which showed that the overall diameter of the aerosol increased from 0.0848 μm at the start of cutting to 0.1247 μm at the end of the cutting. The calculation results were compared with the concentration by diameter over time. During the cutting process, particles with a diameter of 0.06 μm or smaller were continuously measured. In comparison, particles with a diameter of 0.2 μm or larger were found to increase in concentration after a certain time following the start of cutting. In addition, when the aerosol was measured after the cutting process had ended, particles with a diameter of 0.06 μm or less, which were measured during cutting, were hardly detected. These results show that the nucleation-sized aerosols are generated during the cutting process, which can explain the measurement of small particles at the beginning of cutting. In addition, it can be speculated that the generated aerosols undergo a process of growth by contact with the atmosphere.

This study presents the results of real-time aerosol analysis during the plasma arc cutting of stainless steel. This study shows the generation of nucleation-sized particles at the beginning of the cutting process and the subsequent increase in the aerosol particle size over time at the worksite. The analysis results can characterize the size of aerosol particles that workers may inhale during the dismantling of nuclear power plants.

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Keywords: Aerosol, Aerodynamic diameter, HR-ELPI+, Plasma arc cutting, Real-time

Removal and Isolation of Radioactive Cobalt Using DNA Aptamers

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Electricity generation using nuclear power has various advantages, such as carbon reduction, but the treatment of nuclear waste is emerging as a big issue in many countries. The development of technology that can selectively remove radionuclides from liquid radioactive waste is one of the ways to reduce nuclear waste. Here, we assessed a new way of removing radioactive cobalt from a liquid using an aptamer. Aptamers specifically binding cobalt ions were selected through systematic evolution of ligands by exponential enrichment (SELEX). Their binding strength and stability of their complexes with cobalt were analyzed through surface plasmon resonance assay and 2D program Mfold, respectively. The optimal aptamer/bead conjugate conditions for binding cobalt were established using a FA-C1 aptamer with the strongest binding to cobalt. Under these conditions, more than 80% of radioactive cobalt was removed, and more than 99.95% of removed cobalt was recovered. These results proved that radioactive cobalt removal using this aptamer can effectively reduce liquid radioactive waste. This means that the aptamer/bead complex can be utilized to remove various radioactive metal ions.

Keywords: Aptamer, Cobalt, Radioactive wastewater, Radioactive material, Bead

Characteristics Analysis of Organic Decontamination Liquid Waste Generated During Decommissioning of NPPs and Treatment Method Using Conditioning Process

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A large amount of small and medium-sized metal waste is generated during the decommissioning of nuclear power plants (NPPs). Metal waste is mostly contaminated with low-level radioactive, so it needs decontamination for self-disposal and recycling. A large amount of Organic Decontamination Liquid Waste during decontamination will be generated. The generated organic liquid waste is low in concentration, so the decomposition efficiency is low in the decomposition process. A conditioning process is necessary to concentrate at a high concentration. For effective treatment for Organic Decontamination Liquid Waste, the composition of organic liquid waste and conditioning process were analyzed.

Organic acids, metal ions, radioactive nuclides, surfactants, etc. are present in the Organic Decontamination Liquid Waste, and suspended solids are sometimes generated by various reactions. According to previous studies, the concentration of organic acids including surfactants obtained results from several tens of ppm to a maximum of 1,000 ppm, so the maximum value of 1,000 ppm was assumed. For the composition and total amount of metal ions, the average value (52.7wt% Fe, 16.3wt% Ni, 15.1wt% Cr, 15.9wt% Mn) of the distribution of metal species removed by the actual decontamination process is applied, and the total amount is 1,000 ppm was assumed. As for the radionuclides, only ^{60}Co and ^{137}Cs , which are expected to be mainly present, were considered, and ^{60}Co was assumed to be 2,000 Bq/g and ^{137}Cs to be 360 Bq/g by referring to the literature. The amounts of suspended solids were assumed to be 500 ppm by referring to the characteristics of the liquid waste generated in the decontamination process of the NPPs. Based on the estimated value, a reaction formula was established and a simulated Organic Decontamination Liquid Waste was prepared. As a result of measurement using an analysis device, the composition of the estimated and simulated Organic Decontamination Liquid Waste had similar values.

The conditioning and treatment process largely consists of pretreatment, conditioning, decomposition processes. Organic Decontamination Liquid Waste goes through a pretreatment process to remove impurities with large particles. In the conditioning process, treated water that has passed through the UF/RO membrane system is discharged into the environment. At this time, Concentrated water goes through a decomposition process for processing the Organic Decontamination Liquid Waste, and is discharged to the environment through a secondary RO membrane system. The conditioning process is the low-concentration Organic Decontamination Liquid Waste in the UF membrane system is forming a micelles in an RO membrane system, concentrating it to a high concentration and then go through a recirculation process in the UF membrane system. An experiment was conducted to confirm whether the concentration of surfactants occurred during the conditioning process. As a result of the experiment confirmed that the highly concentrated surfactant formed micelles and was filtered out in the UF membrane system.

Acknowledgements

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Keywords: Organic decontamination liquid waste, Conditioning process, Treatment, Concentration

5분과

제염해체(Poster)

Laser Cutting of Stainless Steel Plates Using a Short Focal Length Head for Safe Nuclear Decommissioning

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Laser cutting has many advantages, including high-speed cutting potential, no reaction forces, narrow kerf widths, ease of remote control, and more. This makes it the next generation cutting technology for nuclear decommissioning. For this reason, various groups in countries with nuclear power plants have been working on applying laser cutting to nuclear decommissioning.

Our group has also been developing in-air and underwater laser cutting technologies. Previous research has focused on efficiently cutting thicker steels. To accomplish this, a cutting head with a long focusing element with a focal length of 600 mm was utilized. A long focusing head is advantageous for cutting thick objects at high speeds because it can maintain a high power density over a long distance. However, with such a long focused beam, the residual laser power that remains after passing through the target object can cut or damage other unwanted objects located behind the target. Utilizing a short focused element can solve this problem, but if the focal length is too short, the cutting capability will be reduced.

In this work, we developed and applied a cutting head that utilizes a focused element with a short focal length of 300 mm. Cutting tests with this head allowed us to cut 10-60 mm thick stainless steel plates at a laser power of 6 kW. We also obtained the maximum cutting speed and kerf width for each thickness while increasing the laser power by 1 kW from 1 to 6 kW. The results obtained in this work are expected to be utilized for safe cutting in future nuclear decommissioning applications.

Acknowledgements

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Keywords: Laser cutting, Laser processing, Nuclear decommissioning, Nuclear dismantling

Development of a Laser Cutting-based Aerosol Analysis System for Dismantling of Nuclear Power Plants

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Laser cutting has been recognized as one of key techniques in dismantling nuclear power plants as it has several advantages such as a remote operation and a reduced secondary waste. However, it generates a significant amount of aerosols that can pose a health risk to workers and further induce environmental pollution during the cutting operation. Thus, understanding the aerosol characteristics generated by the laser cutting is crucial for implementing an effective cutting operation and reducing the exposure to these hazardous particles. In this work, we established a methodology to collect the aerosols and investigate their properties in the laser cutting operation. We built an integrated laser cutting system for aerosol analyses, consisting of a high-power laser cutting module, a metal sample holder, an aerosol collector, and a closed chamber. We expect that this system will offer an opportunity for in-depth understanding of the aerosol properties, by connecting it with desired type of aerosol analysis platforms, and further safe dismantling operation of the nuclear power plants.

Acknowledgements

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Keywords: Nuclear dismantling, Nuclear decommissioning, Laser cutting, Aerosol analysis

Considerations on the Response of Unexpected Incidents and Accidents During Decommissioning of Nuclear Facilities

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Despite of careful planning of decommissioning projects, there are often surprises when facilities are opened for dismantling purposes, or when material is removed from hot cells, etc. Unexpected incidents and findings during the decommissioning of nuclear facilities have been referred to in the past as unknowns. However, many of the problems encountered during implementation of decommissioning are well known, it is simply that they were not expected to arise. In some other cases, the problem may not have been encountered in the decommissioning team's experience, forcing the development of new techniques, tools and procedures to address the unexpected problem, with the attendant delays and cost overruns that this often involves. Unknowns in decommissioning cannot be eliminated, regardless of the efforts applied. This is especially the case in old facilities where documentation may have been lost or where modifications were carried out without updates to reports. As a result, when planning for decommissioning, it is prudent to assume that such problems will occur, and ensure that arrangements are in place to deal with them when they arise. This approach will not only improve the efficiency of the decommissioning project, but will also improve the safety of the operations. One of the most common root causes of unexpected events in decommissioning is the lack of detailed design information or missing records of modifications, maintenance issues and incidents during operation. It is therefore necessary to check the completeness of design information in existing plants and to ensure that configuration management techniques are applied at all stages of the lifetime of a plant. In the case of a new plant, archiving samples of materials can be a valuable source of information to support decommissioning planning. During the lifetime of plants, it is likely that modifications will be carried out involving the construction of new buildings. The opportunity should be taken in these circumstances to consider the layout, the physical size and other attributes of the plant to ensure that they do not make decommissioning of existing facilities more difficult and also to optimize the potential for reuse in support of the decommissioning of the whole site, later in the life of the facility. Characterization of all aspects of a plant is essential to reduce the number of unknowns and the likelihood of unexpected events. This characterization should be extensive, but there is a limit to the level of detail that should be sought as the cost versus benefit gain may reduce. Reducing unknowns by retrospectively obtaining physical data associated with a facility is a useful means of characterization, and there are many tools in existence that can be used to carry this out accurately and effectively. Regardless of the efforts that are employed in decommissioning planning, unexpected events should be anticipated and contingency plans prepared. Although the details of the event itself may not be anticipated, its impact may affect safety and environmental discharge, and may or may not involve radiological impacts. Regardless of more serious impacts, unexpected events are likely to result in modifications to the decommissioning plan, incur delays and cost money. Finally, regardless of the status of a facility, whether at the concept stage or at the decommissioning stage of its life cycle, it is never too early to begin thinking and planning for decommissioning.

Keywords: Accident, Decommissioning, Incident, Safety

A Case Study on Clearance of Radioactive Material and Removal of Non-radioactive Material From the Decommissioning NPPs in Germany

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The operation and decommissioning of nuclear power plants (NPPs) creates waste in the process of handling radioactively contaminated material, which must be disposed of in a repository or can be recovered of in the same way as conventional waste in the course of handling radioactively contaminated materials. For buildings or sites of NPPs it also has to be decided under what conditions they can continue to be used for other, conventional purposes or demolished. This decision is referred to as “release from supervision under nuclear and radiation protection law” or “clearance” in short. The clearance levels applicable in Germany according to the Radiation Protection Ordinance have been defined such that a radiation dose (hereinafter referred to as “dose”) of 10 μ Sv per year is not exceeded. The vast majority of the materials resulting from the dismantling of a nuclear power plant (e.g. most of the massive concrete structures) are neither contaminated nor activated. Thus, there is no need to treat these materials as radioactive waste. Emplacement of uncontaminated masses which in Germany is essentially several million tonnes of building rubble in a repository would require additional construction of such facilities, which, in view of the negligible hazard potential, from the point of view of the Nuclear Waste Management Commission (ESK) is clearly to be rejected both economically and, in particular, ecologically. Alternative ways are increasingly discussed in public, such as the abandonment of buildings after gutting, i.e. refraining from demolition of the controlled area buildings of NPPs. Also, another proposal discussed in public, the landfilling or the long-term storage of cleared material at the site, does not offer any safety-related advantages either in the view of the ESK. If, after completion of all dismantling work, the building has been decontaminated such that the clearance levels for buildings are complied with further use of the building rubble resulting from demolition is harmless from a radiological point of view. For these reasons, Germany has deliberately decided to use clearance as an essential measure in the dismantling of NPPs.

If it is intended to conventionally reuse or depose of virtually contaminant-free material from controlled areas, it must first undergo a clearance procedure. The prerequisites that must be fulfilled for clearance are regulated in the Radiation Protection Ordinance, which includes two basic clearance pathways: unrestricted and specific clearance. In the following, the basic process of clearance is briefly presented and illustrated for a better understanding. It comprises five steps. Step 1-Radiological characterization by sampling, Step 2-Dismantling of plant components in the controlled area, Step 3-Decontamination, Step 4-Decission measurements, Step 5-Clearance and further management. The entire clearance process is governed by a clearance notice and is carried out under the supervision of the competent authority under nuclear and radiation protection law or the independent authorized expert commissioned by it. The clearance pathways contained in the Radiation Protection Ordinance have proven themselves in practice. They permit safe and proper management of material from dismantling and release of the site from supervision under nuclear and radiation protection law. These German regulatory procedures should be taken into account and deregulation and removal should be used as appropriate and necessary tools in the process of decommissioning NPPs in order to return non-hazardous materials to the material cycle or for conventional disposal.

Keywords: Decommissioning, 10 μ Sv, Nuclear waste management commission, Unrestricted area

Considerations on the Investigations of Incidents and Accidents Related to Industry Safety During Decommissioning of Nuclear Facilities

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Working during decommissioning of nuclear facilities can subject workers to a number of industrial health and safety risks. Such facilities can contain hazardous processes and materials such as hot steam, harsh chemicals, electricity, pressurized fluids and mechanical hazards. Workers can be exposed to these and other hazards during normal duties (including slips, trips and falls, driving accidents and drowning). Industrial safety accidents, along with their direct impacts on the individuals involved, can negatively affect the image of nuclear facilities and their general acceptance by the public.

Industrial safety is the condition of being protected from physical danger as a result of workplace conditions. Industrial safety program in a nuclear context are the policies and protections put in place to ensure nuclear facility workers are protected from hazards that could cause injury or illness.

Preventive actions are those that detect, preclude or mitigate the degradation of a situation. They can be conducted regularly or periodically, one time in a planned manner, or in a predictive manner based on an observed condition. Corrective actions are those that restore a failed or degraded condition to its desired state based on observation of the failure or degradation.

In industrial safety, the situations or conditions of interest are those observed via the performance monitoring, investigations, audits and management reviews. Preventive and corrective actions are those designed to place or return the system to its desired state. Preventive actions where possible are preferred as they eliminate the adverse condition prior to it occurring.

When an accident or incident occurs, the primary focus is on obtaining appropriate treatment for injured people and securing the scene to prevent additional hazards or injuries. Once the injured personnel have been cared for and the scene has been secured, it is necessary to initiate a formal investigation to determine the extent of the damage, causal factors and corrective actions to be implemented. Certain tools may be needed to investigate such incidents and accidents. Initial identification of evidence immediately following the incident includes a list of people, equipment and materials involved and a recording of environmental factors such as weather, illumination, temperature, noise, ventilation and physical factors such as fatigue and age of the workers. The five Ws (what, who, when, where and why) are useful to remember in investigation of incidents and accidents.

Keywords: Accident, Decommissioning, Incident, Industrial safety, Investigation

Study on Thermal Decomposition Characteristics of Dry Active Wastes From the Nuclear Facilities

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Dry active wastes (DAWs) are a type of combustible radioactive solid waste, which includes decontamination paper, protective clothing, filters, plastic bags, etc. generated from operating nuclear facilities and decommissioning projects. The volume of DAWs could be increased over time, disadvantage to higher disposal costs and space utilization of disposal site. Additionally, incineration methods cannot be applied to DAWs, unlike general environmental waste, due to concerns about air pollution and the release of harmful chemicals with radioactive nuclides into the atmosphere. Recently, KAERI developed an alternative thermochemical process for reducing the volume of DAW, which involves a step-wise approach, including carbonization, chlorination, and solidification. The purpose of this process is to selectively separate the radioactive nuclides from carbonized DAWs that are less than clearance criteria, which can be disposed of as non-radioactive waste.

In this research, we investigated the thermal decomposition characteristics of DAWs using non-isothermal thermogravimetric analysis, which was performed with different categorized wastes and heating conditions. As a result, the cellulose DAWs such as decontamination paper and cotton were thermally decomposed in three or four-step depending on the heating conditions. On the other hand, the hydrocarbon and rubber DAWs such as plastic bags and latex were thermally decomposed in one or two-step. Therefore, it could be suggested the thermochemical treatment conditions that minimize the decomposition of DAWs by controlling the reaction steps, and we will try to apply these results for cellulose type DAWs such as decontamination paper and cotton, which is generated majorly from the nuclear facilities in the future.

Acknowledgements

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Keywords: Dry active waste, Volume reduction, Thermochemical treatment, Non-isothermal TGA, Thermal decomposition

Decontamination of Simulated Fallout-Contaminated Metal Surface Using a Hydrogel Decontamination Medium

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Nuclear weapon generates huge amount of radioactive fallout which is extremely dangerous. The fallout gradually falls to the ground and then covers every surface in city and nature. A hydrogel decontamination medium has been developed to clean the surface polluted by the fallout. The hydrogel is soluble in water so the used hydrogel can be simply removed from the surface by washing. However, significant amount of waste water, containing the radioactive fallout, is generated with this process. In this respect, it is necessary to secure alternative technical options for the used hydrogel recovery. In this study, a steam-suction process was suggested for the used hydrogel recovery. Contaminated stainless steel surface, with fixed simulated fallout particles, was prepared for test. The simulated fallout particles were obtained by high-temperature treatment of a mixture of natural soil, used concrete, and Fe_2O_3 . The hydrogel, composed of poly-vinyl alcohol and borax, was spread onto the contaminated stainless steel surface. The hydrogel was soft at first and it gradually becomes rigid with time. The used hydrogel was recovered by suction with a simultaneous steam spraying to soften the rigid gel. As a result, the clean surface of the stainless steel without the simulated fallout particles was obtained, showing the feasibility of this technique for the used hydrogel recovery.

Keywords: Nuclear weapon, Fallout, Surface decontamination, Hydrogel, Suction, Steam

Thermal Treatment of Latex Gloves in a Closed Vessel for Waste Reduction

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Various dry active wastes (e.g., gloves, wipers, shoes, clothes) are generated during operation and maintenance of nuclear facilities. Among those, latex gloves get interest because they contain both organic and inorganic compounds. CaCO_3 is a common filler material for production of latex rubbers. Here, latex gloves were thermally treated in a closed vessel to separate the organic and inorganic compounds. Using the closed vessel is beneficial as it can prevent escape of any species, including radioactive nuclides in a real case, generated during the treatment. It was found that thermal decomposition of latex gloves occurred above 250°C . Latex gloves were decomposed to gas, liquid, and solid compounds. The gas product is thought to be volatile organic compounds (VOCs). The liquid product seems to be a mixture of oils and water. A CaCO_3 phase was identified in the solid product, as expected. The VOCs can be easily separated at room temperature by purging in vacuum or inert atmosphere. The liquid-solid mixture can be separated by distillation. It is thought that gamma-emitting nuclides, such as Cs-137, Sr-90, and Co-60, dominantly remain in the solid product. In the best situation, the solid product is the only subject to be transferred to final wasteform fabrication stream and thus volume of final waste can be reduced. Surrogates of contaminated latex gloves (containing Cs, Sr, and Co) were prepared and they were treated at 350°C in the closed vessel. How these contaminants behave in this thermal process will be discussed in the presentation.

Keywords: Dry active wastes, Latex, Thermal decomposition, Decontamination, Waste reduction

~~A Study on the Blasting Decontamination for Surface Contaminated Metallic Waste~~

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Withdrawal

A Study on the Clearance Waste Disposal Measures for Decommissioning of Nuclear Power Plant

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The concept of clearance is to manage radioactive waste by incineration, reclamation, or recycling as non-radioactive waste, excluding those found to have a concentration of less than the allowable concentration of clearance.

Among the types of waste subject to clearance, concrete is managed by recycling and landfill, metal by recycling and reuse, combustible materials by incineration, and soil by landfill.

In Korea, clearance has been implemented in earnest since 2000, and the types and quantity of waste subject to clearance are increasing. For clearance, the nuclear-related operator submits its clearance plan to the regulatory body, and the regulatory body reviews the clearance plan and notifies the operator of its suitability.

Since a significant amount of radioactive waste generated when decommissioning nuclear power plants is expected to be classified as clearance waste, this study will present clearance waste disposal measures for nuclear power plant through a review of overseas cases related to clearance.

Keywords: Decommissioning, Radioactive waste, Clearance, Recycling, Reuse

A Study on the Treatment of Radioactive Waste for Decommissioning of Nuclear Power Plant

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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. In addition, according to the level of radioactivity, it can be divided into high level, intermediate level, low level, and clearance level waste.

In the case of solid radioactive waste, it is necessary to secure disposal suitability in order to deliver it to a disposal facility, so safe and efficient treatment of a large amount of radioactive waste generated during decommissioning is one of the most important issues. For the treatment of radioactive waste generated during decommissioning, technologies in various fields such as cutting, decontamination, melting, measurement, and packaging are required.

Therefore, this study intends to present and application plan for decommissioning domestic nuclear power plants through overseas case studies for the treatment of radioactive waste expected to occur during nuclear power plant decommissioning.

Keywords: Decommissioning, Radioactive waste, Decontamination, Packaging, Radioactivity

Prediction of Distribution Coefficient of Cesium Using Machine Learning Models

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Radioactive contaminants, such as ^{137}Cs , are a significant concern for long-term storage of nuclear waste. Migration and retention of these contaminants in various environmental media can pose a risk to the surrounding environment. The distribution coefficient (K_d) is a critical parameter for assessing the behavior of these contaminants and can introduce significant errors in predicting migration and remediation options. Accurate prediction of K_d values is essential to assess the behavior of radioactive contaminants and to ensure environmental safety. In this study, we present machine learning models based on the Japan Atomic Energy Agency Sorption Database (JAEA-SDB) to predict K_d values for Cs in soils. We used three different machine learning models, namely the random forest (RF), artificial neural network (ANN), and convolutional neural network (CNN), to predict K_d values. The models were trained on 14 input variables from the JAEA-SDB, including factors such as Cs concentration, solid phase properties, and solution conditions which are preprocessed by normalization and log transformation. We evaluated the performance of our models using the coefficient of determination (R^2) value. The RF, ANN, and CNN models achieved R^2 values of over 0.97, 0.86, and 0.88, respectively. Additionally, we analyzed the variable importance of RF using out-of-bag (OOB) and CNN with an attention module. Our results showed that the initial radionuclide concentration and properties of solid phase were important variables for K_d prediction. Our machine learning models provide accurate predictions of K_d values for different soil conditions. The K_d values predicted by our models can be used to assess the behavior of radioactive contaminants in various environmental media. This can help in predicting the potential migration and retention of contaminants in soils and the selection of appropriate site remediation options. Our study provides a reliable and efficient method for predicting K_d values that can be used in environmental risk assessment and waste management.

Acknowledgements

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Keywords: Cesium, Distribution coefficient, JAEA-SDB, Machine learning, Sorption, Variable importance

Corrosion Inhibitors for PHWR Components in HyBRID Decontamination Solution

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KAERI has been developing a new decontamination process that does not contain any organic chemicals in the decontamination solution and minimizes the use of ion exchange resin in the solution as a purifying step. The process is hydrazine based reductive metal ion decontamination for decommissioning (HyBRID) and consists of N_2H_4 , H_2SO_4 and Cu^+ ions.

The primary system of the LWR is composed of materials with high corrosion resistance, such as stainless steel and Inconel, but among the materials, the feeder and header of the primary system of the PHWR are composed of carbon steel (SA106Gr.B) with low corrosion resistance. Therefore, when decontamination of PHWRs, attention should be paid to corrosion of carbon steel. Since Fe_3O_4 , a contaminating oxide film formed on the surface of carbon steel dissolves faster than ferrite or chromite formed on the surface of Inconel or stainless steel, the base material is exposed to the solution and is corroded during decontamination. When a large amount of iron ions is eluted into the decontamination agent due to corrosion of carbon steel, not only the soundness of the base metal is adversely affected, but also the amount of decontamination waste increases.

The purpose of this study is to develop inhibitors that can minimize corrosion of carbon steel when decontamination of PHWRs using the HyBRID decontamination process. CG, 570S and PP3 were selected as corrosion inhibitors. In addition, corrosion tests of carbon steel were conducted in the HyBRID solution with corrosion inhibitors. The best corrosion inhibitors and optimal operating conditions were selected, and HyBRID decontamination agents with corrosion inhibitors were much better in corrosion resistance than existing commercial decontamination agents.

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Keywords: Decontamination, Corrosion, Inhibitor, Carbon steel, PHWR, SA106Gr.B

A Review of Kori Unit 1 Decommissioning Status in View of the FDP

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Kori Unit 1 is about 600MW Pressurized Light Water Reactor as WH type. KHNP got an approval for construction and operation of Kori unit 1 on May 31, 1972 and started commercial operation from Apr. 29, 1987. And then, it was decided to permanently suspend it on Jun. 18, 2017 after 40 years of commercial operation. The Nuclear Safety Act stipulates that if a commercial nuclear power plant is permanently suspended, the utility must submit a Final Decommissioning Plan (FDP) within 5 years. So, KHNP, the utility, developed a FDP for Kori Unit 1 and submitted it to the government in May 2021. In South Korea, the FDP is to be prepared in accordance with the relevant notices and consists of 11 major chapters such as (1) Decommissioning Plan Overview, (2) Project management, (3) Status of Site and Environmental, (4) Decommissioning Strategies and Method, (5) Ease of Decom. Design characteristic, (6) Safety Analysis, (7) Radiation Protection, (8) Decontamination and Dismantling, (9) Radioactive Waste Management, (10) Environmental Impact Analysis, (11) Fire Protection and (12, 13) Etc., References and Glossary.

KHNP has prepared a strategy and system consisting of three areas such as R&D, Engineering and licensing document development to prepare the final decommissioning plan for Kori Unit 1. The promotion system for the preparation of the FDP for Kori Unit 1 is composed of Engineering (HAS Characterization, Dismantling Safety Evaluation, Radiological Environmental Report, Radioactive Waste Treatment and Facility Construction), R&D(KHNP R&D Results such as Process/Work Package /Cost Estimation, Safety Analysis, Contamination and Exposure, Guide for Detailed Characteristic, Site Safety Analysis, RV & RVI Dismantling Process, etc.), Overseas case lessons learned(Taiwan unit 1 NPP FDP and Spain Zorita NPP FDP analysis) and Development of Licensing Document.

KHNP completed the initial completion of the Final Decommissioning Plan for Kori Unit 1 in March 2020 and carried out collecting residents' opinions through public hearings. And then, KHNP supplemented the results of the residents' opinions and applied for license to the Nuclear Safety and Security Commission in May, 2021. Now, KHNP are responding to the FDP licensing review.

Keywords: Kori unit 1, Final decommissioning plan, Nuclear power plant decommissioning, Dismantling process

A Review of Wolsong Unit 1 Decommissioning Status in View of the FDP

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Wolsong Unit 1 is about 679 MW Pressurized Heavy Water Reactor (PHWR). Canada AECL was responsible for Nuclear Steam Supply System (NSSS) design and supply. Wolsong Unit 1 was operated from 1983 to 2019. Currently, Wolsong Unit 1 is under safety management after permanent shutdown. Wolsong unit 1, a heavy water reactor, has the following characteristics.

- Unlike Light Water Reactor, vertical reactors, Heavy Water Reactor is installed horizontally.
- The internal structure of the reactor is more complex than that of a light water reactor (380 pressure tubes in reactor as called Calandria)
- The Calandria Vault, a large concrete structure filled with light water, is located outside of Calandria

In the case of the decommissioning plan of PHWR in Canada, they have adopted a deferred decommissioning strategy that decommissioning begins after permanent shutdown and long-term safety management (30 to 40 years). So, Decommissioning of PHWR in Canada is expected to start in the 2050s.

Nuclear Safety Act stipulates that if a commercial nuclear power plant is permanently suspended, the utility must submit a Final Decommissioning Plan (FDP) within 5 years. So, KHNP, the utility, is developing the FDP for Wolsong Unit 1 and have a plan to submit it to the government by the end of 2024. And then licensing review is expected to take at least two years. The key milestone for decommissioning project has a plan to start decommissioning in 2027 and complete it by 2034, but this is flexible depending of the government's approval for decommissioning and the completion of prerequisites such as spent fuel transfer, etc.

KHNP has prepared a strategy and system consisting of three areas such as R&D, Engineering and licensing document development to prepare the final decommissioning plan for Wolsong Unit 1. The promotion system for the preparation of the FDP for Wolsong Unit 1 is consisted of Engineering (HAS Characterization, Process/Work Package/Cost Estimation, Dismantling Safety Evaluation, Radiological Environmental Report, Radioactive Waste Treatment and Facility Construction), R&D (COG cooperation, KHNP R&D Results), Kori unit 1 lessons learned, etc.

KHNP have the plan that the FDP Draft development by the end of 2023, reflecting engineering services results, R&D results, COG technical cooperation results and lessoned learned on Kori Unit 1. After collecting opinions from residents through a public hearing, the FDP will be submitted to the government by the end of 2024.

It is expected that there will be many difficulties in the development process as it is the world's first FDP development for the commercial Pressurized Heavy Water Reactors.

Keywords: Wolsong unit 1, Final decommissioning plan, Nuclear power plant decommissioning, Dismantling process

Analysis of Size Distribution for Radioactive Aerosols Generated When Decommissioning Nuclear Power Plants

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Metal waste generated during the dismantling of a nuclear power plant can be contaminated with radionuclides. In general, the internal structure is very complex. Thus, metal waste requires various cutting processes. When metal waste is cut, aerosols are generated. Aerosols are generally various particles of very small size suspended in the working area and remain for a considerable period. This may cause internal exposure of workers due to inhalation of radioactive aerosols generated when cutting radioactive metal waste.

This study investigated various cutting processes and the size distribution of aerosols generated during the cutting process. The cutting process is normally classified into thermal cutting, mechanical cutting, and laser cutting. Thermal cutting includes plasma, flame, and oxygen cutting. Mechanical cutting includes mechanical saws, cutters, nibblers, and abrasive water jets. Stainless steel, carbon steel, aluminum, and copper are commonly used as cutting materials in nuclear power plants. The size of the aerosol generated from cutting showed a very diverse distribution depending on the cutting methods and cutting materials. In general, aerosol size is distributed within 0.1-1 μm . This size distribution is different from the 5 μm aerosol size suggested by the ICRP Publication 66 Lung model. These results show that it is necessary to conduct further studies on the size of aerosols generated when decommissioning nuclear power plants.

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Keywords: Radioactive aerosol, Cutting process, Particle size, Metal waste

Development of Concrete Surface Removal Technology Using an Integrated Laser Scabbling System Designed for Decontamination of Highly Radioactive Concrete Facilities

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Laser scabbling has the potential to be a valuable technique capable of effectively decontaminating highly radioactive concrete surface at nuclear decommissioning sites. Laser scabbling tool using an optical fiber has a merits of remote operation at a long range, which provides further safety for workers at nuclear decommissioning sites. Furthermore, there is no reaction force and low secondary waste generation, which reduces waste disposal costs. In this study, an integrated decontamination system with laser scabbling tool was employed to test the removal performance of the concrete surface. The integrated decontamination system consisted of a fiber laser, remote controllable mobile cart, and a debris collector device. The mobile cart controlled the translation speed and position of the optical head coupled with 20 m long process fiber. A 5 kW high-powered laser beam emitted from the optical head impacted the concrete block with dimensions of 300 mm × 300 mm × 80 mm to induce explosive spalling on its surface. The concrete debris generated from the spalling process were collected along the flexible tube connected with collector device. We used a three-dimensional scanner device to measure the removed volume and depth profile.

Acknowledgements

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Keywords: Laser scabbling, Fiber laser, Concrete decontamination, Explosive spalling

The Evaluation of Oxide Removal Rate on Stainless Steel Depending on the Temperature for Full System Decontamination of PHWR

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It is reported that 48 pressurized heavy water reactors (PHWRs) are in operation, and 10 PHWRs including Wolsong-1 NPP have been permanently shut down in the world. In the case of PHWRs, which have been permanently ceased, they are managed through the delayed decommissioning method, but there are no cases of dismantling. Therefore, technology development is urgent for the effective decommissioning of PHWRs. Unlike PWRs, PHWRs are separated into coolant system and moderator system. Most of pipes and systems of coolant system are mainly composed of carbon steel, except of the steam generator tubes which are composed of nickel alloy. On the other hand, the moderator system is composed of stainless steel. In the case of stainless steel, the inner layer of the oxide film is composed of chromium oxide, and the outer layer is composed of iron and nickel oxide in enriched. To remove two oxide layers, it is needs to different decontamination method, the coolant system can perform the system decontamination process through a reduction process, but in the case of the moderator system, the oxidation/reduction process is required because it has a material and oxide film similar to PWRs.

In this study, this is evaluated the oxide film removal rate according to the type of stainless steel and temperature in order to remove the oxide film deposited in the moderator system. The experiments were carried out at temperatures of 60, 70, 80 and 90°C, with a concentration of 200 ppm of permanganic acid and nitric acid, and 2,000 ppm of oxalic acid, respectively. The results of the oxide film removal rate test for SUS304 showed 29% at 60°C, 38% at 70 and 80°C, and 41% at 90°C. For SUS403, the oxide film removal rate experiment results showed 62% at 60°C, 85% at 70°C, 94% at 80°C, over 99% at 90°C. The results showed that the removal efficiency of the oxide film increased as the temperature increased. Following the results of experimental, the optimum temperature of oxide removal in composed of the stainless steel material is to be 90°C for decontamination of PHWR.

Keywords: Stainless steel (SUS304, SUS403), Oxide film, Permanganic acid, Oxalic acid, PHWR, Decontamination

The Study of Organic Acid Decomposition Depending on UV Lamp Contact Time for Reducing Waste of Decontamination

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Radioactive products generated by long-term operation at NPP can become deposited on the surfaces of the system and equipment, leading to radiation exposure for workers during the decommissioning process. Chemical decontamination is one of the methods to reduce radiation exposure of workers, and there are HP CORD UV, CITROX, CAN-DECON. In the chemical decontamination process, organic acids are generally used, and representative organic acids include oxalic acid and citric acid. There are various methods for removing residual organic acid in decontamination liquid waste, such as using an oxidizing agent and an ion exchange methods. However, there is a problem in that oxidizing agent is used excessively or secondary wastes are generated in excess during organic waste treatment. However, when organic acid is decomposed using a UV lamp, the amount of secondary waste is reduced because it is decomposed into CO_2 and H_2O .

In this study, organic acid decomposition was evaluated as the contact time of the UV lamp. The experimental equipment consists of a UV reactor, a mixing tank, a circulation pump. The experimental conditions involved preparing 60 L of organic liquid waste containing oxalic acid, hydrogen peroxide and iron chloride. Test A was conducted using one UV reactor, and Test B was performed by connecting two UV reactors in series. As a result of the experiment, a decomposition rate of over 95% was shown after one hour for oxalic acid, and it was confirmed that the initial decomposition rate was faster as the contact time increases. Therefore, in order to increase the initial decomposition rate, it is necessary to increase the contact time of the UV lamp by connecting the UV reactors in series.

Keywords: UV reactor, Oxalic acid, Hydrogen peroxide, Contact time

Distribution Coefficient of Cesium and Strontium on Various Clay Minerals in the Presence of Seawater

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The mobility of radionuclides is largely determined by their radiological properties, geochemical conditions, and adsorption reactions, such as surface adsorption, chemical precipitation, and ion exchange. To evaluate the safety assessments of radionuclides in nuclear sites, it is essential to understand the behavior and mechanism of radionuclides onto soils. Since nuclear power plants are located in coastal areas, the chemical composition of groundwater can vary depending on the intrusion of seawater, altering the adsorption distribution coefficient (K_d) values of radionuclides. This study examines the impact of seawater on the K_d values of clay minerals for cesium (Cs) and strontium (Sr). The results of Cs^+ adsorption experiments showed a broad range of K_d values from 36 to 1,820 mL/g at an initial concentration of 1 mg/L and a high sorption coefficient of 15-613 mL/g at an initial concentration of 10 mg/L. Montmorillonite, hydrobiotite, illite, and kaolinite were ranked in terms of their CEC values for Cs^+ adsorption, with hydrobiotite having the highest adsorption at 1 mg/L. The results of Sr adsorption experiments showed a wide range of K_d values from 82 to 1,209 mL/g at an initial concentration of 1 mg/L and a lower adsorption coefficient of 6.68-344 mL/g at an initial concentration of 10 mg/L. Both Cs^+ and Sr^{2+} demonstrated lower K_d values at higher initial concentrations. CEC of clays found to have a significant impact on Sr^{2+} K_d values. Ca^{2+} ions showed a significant impact on Sr^{2+} adsorption distribution coefficients, demonstrating the greater impact of seawater on Sr^{2+} compared to Cs^+ . These findings can inform future safety assessments of radionuclides in nuclear sites.

Acknowledgements

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Keywords: Cesium, Strontium, Clay mineral, Distribution coefficient, Cation exchange capacity (CEC)

Exploring the Potential of Zeolites for Selective Separation of Hydrogen Isotopes

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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. This study have the literature review for previous research on D_2/H_2 adsorption and analyzes the D_2/H_2 adsorption behaviors of hydrogen isotopes for various zeolite using BET at 77 K. The study predicts the D_2/H_2 adsorption selectivity based on the results obtained with BET. These hydrogen isotope adsorption fundamentals provide a foundation for future processes for tritium separation.

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Keywords: Hydrogen, Isotope, Separation, Zeolite

Copper Based Metal-Organic Frameworks for Efficient Photocatalytic Degradation of Organic Dyes

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The removal of aqueous pollutants, including dye molecules from wastewater remains one of the pressing problems in the world. Because of chemical stability and conjugated structure, dye molecules cannot be easily decomposed by heat with oxidizing reagents such as H_2O_2 and light. The most common representative of widespread organic pollutant is methylene blue (MB) with molecular formula $\text{C}_{16}\text{H}_{18}\text{ClN}_3\text{S}$, which is important colorant and used in various chemical and biological production industries and causes serious environment problems. Porous materials, including MOFs (metal-organic frameworks) have been applied for efficient MB photocatalytic degradation. However, one of the main barriers to using most MOFs to break down aromatic organics is wide band gap energy, which means that the catalyst can exhibit high photocatalytic performance only under UV-light irradiation. Moreover, most MOFs usually show the poor water stability of frameworks, which tend to dissolve in water with total destruction.

In this work we report about two new copper based MOFs with high photocatalytic properties for efficient MB degradation from wastewater under UV-light and natural sunlight. Time, required for 100% MB degradation, equals 7 minutes under UV (source 4 W 254 nm VL-4.LC UV-lamp) and 60 minutes under natural sunlight irradiation in the presence of H_2O_2 . Crystal structure information is provided using single crystal X-ray diffraction data. The composition and comparative characteristics of MOFs are given using powder X-ray diffraction, UV-visible diffuse reflectance spectroscopy, UV-visible spectroscopy and Fourier-transform infrared spectroscopy.

Keywords: Metal-organic framework, Methylene blue degradation, Photocatalyst

Removal of Radioactive Concrete by a High-Power Fiber Laser Scabbling

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Concrete decontamination tools capable of removing the nuclear contaminated surface are necessary to minimize the amount of concrete waste generated in the process of decontamination and dismantling of nuclear power plants. Laser scabbling is a decontamination technique that removes the contaminated surface layers concrete surface by inducing internal explosion. The application principle of laser scabbling technology uses the porous nature of concrete including moisture. When high thermal energy is applied to the concrete surface, an explosion at pores is induced along with an increase in water vapor pressure. High-powered laser beam can be an effective induction source of local explosive spalling on concrete surface. In this study, the scabbling test using a 5 kW high-powered fiber laser was conducted on the concrete blocks to establish the optimal conditions for surface decontamination. It was also measured the volume peeled off the concrete surface under the conditions of two different laser head speeds. Furthermore, we tested the removal efficiency of radioactive concrete particles generated during high-power fiber laser scabbling process.

A 5 kW laser beam was applied to the concrete surface at two different laser head speeds - 120 mm/min and 600 mm/min. The laser beam repeatedly moved 200 mm horizontally and 40 mm vertically within the concrete block. The amount of surface concrete removed from concrete block was calculated from the measurement of the volume and mean depth using a 3D scanner device (laser-probed Global Advantage 9.12.8(HEXAGON)) for the two different the laser head speeds. By increasing the laser head speed, less explosive spalling occurred due to shorter contact time of the laser beam with the concrete. The laser head speed of 600 mm/min reduced about 89% of the waste generated by shallow depth of scabbling as compared to the waste generated at the laser head speed of 120 mm/min.

The fiber laser scabbling system was developed for surface decontamination of radioactive concrete in nuclear power plants. Tests were performed to find the optimum parameters to reduce the generation of particulate waste from the contaminated concrete surface by controlling the laser head speeds. It was confirmed that the wastes from surface decontamination was reduced up to 89% by increasing laser head speed from 120 mm/min to 600 mm/min. It was also observed that the cylindrical tube effectively vacuumed the debris generated by the explosive spalling into the collector. Removal efficiencies of concrete particles were measured greater than 99.9% with ring blower power of 650 air watt of the filter system.

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Keywords: Nuclear decommissioning, Radioactive concrete, Surface decontamination, Fiber laser, Scabbling

Atacamite as a Hydrogen Isotope Fixation Mineral

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Heavy water (D₂O) is a coolant as well as a moderator of pressurized heavy water reactors (PHWRs). During operation of PHWRs, deuterium (H-2, D) in heavy water is gradually converted to tritium (H-3, T), which is a radioactive nuclide with a half-life of 12.3 years, by capturing neutron. Various radioactive wastes contaminated by T are generated upon the PHWR operation. Owing to the similarity of D and T, they behave together a form of water (either liquid or vapor) in a normal circumstance. To handle D and T with the water form is quite difficult because it is not a solid and is highly mobile in nature. In this study, a mineralization technique to fix D and T in a solid form is suggested. It is considered that hydroxide minerals, which have low solubility in water, might tightly bind D and T in non-mobile, solid-state media. Feasibility of this strategy is studied by using a copper-based hydroxide mineral, atacamite. Atacamite is a natural mineral found in copper deposits with chemical formula of Cu₂Cl(OH)₃. Atacamite can be simply synthesized in laboratories by a precipitation method using copper chloride and calcium carbonate as precursors. Both chemicals were added into heavy water to obtain pale-green precipitates. Heavy water is the only source for D in this reaction and thus deuterated mineral is expected to be form. The obtained deuterated mineral, suspected to be Cu₂Cl(OD)₃, was then immersed in natural deionized water (extremely low D₂O concentration) for several days to identify how fast D in Cu₂Cl(OD)₃ dissolves into water. In a preliminary Fourier transform infrared (FTIR) spectroscopy, absorption peaks related to HDO and D₂O were not observed in the deionized water which is recovered after the immersion test, suggesting that D remained stable in the synthesized mineral. However, owing to low detection limit of FTIR, more precise analysis should be taken to clearly identify the stability of D of the deuterated atacamite. If deuterated hydroxide minerals are found to have sufficiently high D stability in natural water, they can be further treated with cement or other stabilization media to form a final wasteform for underground disposal.

Keywords: Deuterium, Heavy water, Tritium, Tritiated water, Fixation, Hydroxide mineral, Atacamite

A Study on Disposal Safety Analysis of Cellulose Containing Radioactive Waste

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Dry active waste (DAW) contains substantial amount of cellulose related materials. The DAW are usually classified as low and/or very low-level waste. In Korea, three types of disposal facilities have been considered: silo, engineering barrier, and land-fill. Currently, only the silo type disposal facility is in operation. Around 27 thousand drums were disposed in silo. Massive amount of cement concrete is used in construction of silo. The ground waste, which flow through the concrete structure, shows higher pH than as it is. It is generally known that the pH of silo is ~ 12.47 in Korea, when considering construction material, filling material, and property of ground water. It is expected that the cellulose in DAW will be partially transformed to isosaccharinic acid (ISA). It is generally accepted that the ISA plays a negative role in safety analysis of disposal facility by stimulation of specific nuclides. Various factors affect the degradation of cellulose containing radioactive waste, such as degree of polymerization, pH of disposal condition, interaction between concrete structure and ground water, etc.

In this paper, the disposal safety analysis of cellulose containing radioactive, usually paper, cotton, wood, etc., are studied. The degradation of cellulose with respect to degree of polymerization, pH of neighboring water, filling material of silo, etc. are reviewed. Based on the review results, it is reasonable to conclude that the substantial amount of DAW could be disposed in silo.

Keywords: Radioactive waste, Cellulose, Disposal, Safety analysis

A Study on Nuclide Characteristics in Activation Analysis of Activated Components

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The segmentation of activated components is considered as a one of the most important processes in decommissioning. The activated components, such as reactor vessel and reactor vessel internals, are exposed to neutron from the nuclear fuel and classified to intermediate, low, and very low-level wastes. As it is expected, the components, which are closed to nuclear fuel, exhibit higher degree of specific activity.

After the materials were exposed to neutrons, their original elements transform to other nuclides. The primary nuclides in activated stainless steel are ^{55}Fe , $^{63,59}\text{Ni}$, ^{60}Co , ^{54}Mn , etc. The previous study indicates that the specific activity of individual nuclide is strongly depends on the material compositions and impurities of the original materials. The ^{59}Co is the one of the most important impurities in stainless steel and carbon steel.

In this paper, the relationship between individual nuclides in activation analysis of activated components was studied. The systematic study on specific activity of primary nuclides will be discussed in this paper to understand the activation tendency of the components.

Keywords: Activation analysis, Reactor vessel, Reactor vessel internals, Radioactive waste

A Study on Management of Non-Fuel Radioactive Waste in Spent Fuel Pool

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During the operation of the nuclear power plant, various radioactive waste are generated. The spent resin, boron concentrates, and DAW are classified as a generic radioactive waste. They are treated and stored at radioactive waste building. In the reactor vessel, different types of radioactive waste are generated. Since the materials used in reactor core region exposed to high concentration of neutrons, they exhibit higher level of surface dose rate and specific activity. And they are usually stored in spent fuel pool with spent fuel.

Various non-fuel radioactive wastes are stored in spent fuel pool, which are skeleton, control rod assembly, burnable neutron absorber, neutron source, in core detector, etc. The skeleton is composed of stainless 304 and Inconel-718. There are two types of control rod assembly, that are WH type and OPR type. The WH type control rod is composed of Ag-In-Cd composites. The OPR type control rod is composed of B₄C and Inconel-625.

In this paper, the characteristics and storage status of the non-fuel radioactive waste will be reported. Also, the management strategy for the various non-fuel radioactive waste will be discussed.

Keywords: Non-Fuel radioactive waste, Skeleton, Control rod, Burnable neutron absorber, Secondary neutron source

A Study on Characterization of Large Concrete Drum Using Non-Destructive Method

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The large rectangular and cylindrical concrete drums are stored in nuclear power plant (NPP) for a long time. At the early stage of NPP operation, the treatment technology of boron concentrates and spent resin was not well developed, when compared to current system. Since the waste acceptance criteria (WAC) of the disposal facility was not established, the boron concentrates and spent resins were packaged in 200 L drum. Some of the 200 L drums, which contain relatively high dose rate radioactive waste, were stored in large concrete drum. The concrete drum offers superior shielding effect and allows reduction of radiation exposure to workers. The WAC requires various characteristics: radiological characteristics, physical characteristics, chemical characteristics, etc. The non-destructive method allows the rapid evaluation and estimation of the concrete structure. Also, it is expected that the large concrete exhibits integrity after the measurements.

In this paper, the non-destructive method to understand the large rectangular and cylindrical drum is systematically studied. The advantage and disadvantage of the non-destructive methods were compared in this paper. In addition, the optimized methodology to characterize the radioactive waste containing large rectangular and cylindrical drum will be discussed in this paper.

Keywords: Large concrete drum, Waste acceptance criteria, Non-destructive method, Radioactive waste

Status Update for Facilities About Decommissioning of Fukushima Daiichi NPP

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On March 11 2011, Fukushima Daiichi nuclear power plant site was attacked by a huge tsunami caused by Tohoku Pacific Ocean earthquake. Nuclear fuels of unit 1, 2, and 3 of Fukushima Daiichi nuclear power plant was melted down by the disaster. After the accident, Japan's government has announced "Mid-and-Long-Term Roadmap towards the decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4". The topics of roadmap is made of measures to deal with contaminated water, removal of fuel rod assemblies from spent fuel pools, retrieval of fuel debris, measures to deal with waste materials, and other operations. To support the activity of the roadmap, various facilities about decommissioning have been established and operated on inside or outside of Fukushima Daiichi nuclear power plant site. Representatively, Collaborative Laboratories for Advanced Decommissioning Science which conducts R&D decommissioning, Naraha Remote Technology Development Center which develops remotes robots and VR (Virtual reality), Okuma Analysis and Research Center which performs radiochemical analyses for radioactive waste, and Fukushima Environmental Safety Center which conducts environmental dynamics and radiation monitoring.

Keywords: Fukushima daiichi nuclear power plant, Roadmap, Decommissioning, Facilities about decommissioning

Current Status of Treatment Method for HWR Resin Waste and C-14 Recovery Technology in Resin Waste

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The intermediate level spent resins waste generated from water purification for the the moderator and primary heat transport system during operaioin of heavy water reactor (HWR). Especially, moderator resins contain high level activity largely because of their C-14 content. So spent resins are considered as a problematirc solid waste and require special treatment to meet the waste acceptance criteria for a disposal site. Various methods have been studied for the treatment of spent resins which include thermal, destructive, and stripping methods. In the case of solidification methods, cement, bitument or organic polymers were suggested. In the 1990s, acid stripping using nitric acid and thermal treatment methods were actively investigated in Canada to remove C-14 nuclide from waste resin. In Japan, thermal distructive method was studied in the 1990s. Since 2005, KAERI developed acid stripping method using phosphate salt. However, acid stripping method are not suitable due to large amounts of 2nd waste containing acid solution with various nuclides. To solve this probelm, KAERI has been suggested the microwave treatment method for C-14 selective removal from waste resin in the 2010s. Pilot scale demonstration tests using radioactive waste resin generated from Wolsung unit 1 and unit 2 were successfully conducted and 95% of C-14 was selectively removed from the radioactive waste resin.

In recent years, price of C-14 source is dramatically increased due to market growth of C-14 utilization and exclusive supply chain depending on China and Russia. High purity of C-14 were captured in HWR waste resin. Interest of C-14 recovery research from HWR waste resin is currently increased in Canada.

In this study, microwave method is suggested to treat HWR waste resin with C-14 recovery process. Additionally, status of waste resin management and research trends of HWR waste resin treatment are introduced.

Keywords: HWR, Waste resin, Microwave, C-14, Recycling

Development of Rotary Filtration Membrane Equipment for Radioactive Sludge Recovery

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Various types of tanks are used in nuclear power plants, and sludge composed of various organic substances and inorganic oxides contaminated with radioactive materials may be present at the bottom of a tank of a radioactive waste treatment device. In addition, glassy and fixative oxide contamination layers are accumulated on the inner wall of the tank depending on the tank material, usage and degree of oxidation.

Such contaminated sludge is the main cause of radiation exposure to workers when dismantling nuclear power plant tanks. In addition, the waste filters generated by filtration of contaminated sludge is treated as secondary radioactive waste, and this radioactive waste not only occupies a lot of disposal space, but also the disposal cost is continuously increasing. Therefore, it is necessary to develop a technology that does not generate waste filters as much as possible.

To solve this problem, NILEPLANT Co., Ltd. registered a patent named “Filtering apparatus” based on previous research and manufactured a rotary filtration membrane device through detailed design.

The rotary filtration membrane device is composed of three or more multiple rotary filtration membranes, and can remove fine particles in wastewater as well as sludge accumulated inside a radioactive contamination tank. In addition, considering the site characteristics of special conditions such as nuclear power plants, it was designed to show excellent performance in removing fine particles while minimizing the area where the device is installed.

The rotary filtration membrane device is designed and manufactured as a double cylinder structure that combines a hydro cyclone filter type body and an inner partition wall, and is equipped with a filter cloth-based rotary cylinder filter to process sludge through the filter cloth in addition to inertial. In addition, the patented principle enables self-backwashing without stopping the filtration process, extending the life of the filter and minimizing waste filters.

The filtration performance, self-backwashing function, and sludge behavior of the rotary filtration membrane device manufactured based on the detailed design were evaluated through experiments, and improvements to obtain more effective filtration performance were derived. Accordingly, it is expected that the more improved rotary filtration membrane device can be effectively used to remove sludge generated during the dismantling of nuclear power plants in the future.

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Keywords: Nuclear power plant decommissioning, Dismantling radioactively contaminated tanks, Radioactive sludge, Rotary filtration, Rotary filtration membrane

A Study on the Status of Radioactive Effluent Release After Permanent Shutdown of PWR Nuclear Power Plants

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In case of Korea, unlike overseas nuclear power plants, adjacent units are located in permanently stopped nuclear power plants. Radioactive substances from airborne and liquid effluents are released into the environment from the NPP, and the radioactivity of the released substances must be reported to the regulatory authorities. Radioactive effluents are released into the environment not only in operation but also after permanent shutdown. Due to domestic conditions in which multiple units exist on the same site, it is necessary to consider radioactive effluents generated after permanent shutdown of NPPs.

In particular, liquid effluent may have an increased tritium concentration due to draining the spent fuel pool. This paper summarizes the annual liquid emissions of PWR power plants that have been permanently shut down. The data was obtained from the Nuclear Regulatory Commission's (NRC) annual radioactive effluent release report, which provides information on the annual emissions power plants into the environment. The liquid emissions of each plant were organized into an annual table, providing an overview of the amount of liquid released by each plant.

This study aims to raise awareness about the potential environmental impact of permanently shut down nuclear power plants and the need for proper management of their liquid emissions. The findings of this study can be used by operator, policymakers, and other stakeholders to make informed decisions regarding the decommissioning and management of nuclear power plants.

Keywords: Tritium, Liquid effluent release, Decommissioning

Technology Development of Core Sampling for Evaluation of Degradation Performance of Radiologically Contaminated Concrete

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The concrete structure of a nuclear power plant is a major safety structure that performs shielding functions to block radioactive materials and radiation, heat removal, and isolation functions. Therefore, concrete structures of nuclear power plants must prove structural safety from immediately after construction to dismantling, and a representative method for this is to investigate compressive strength. The compressive strength and specimen standards of concrete structures are specified in ASTM C 42/C 42M, and samples must be obtained through core drilling in order to collect samples according to this standard. However, commercial equipment requires anchor installation work causes radiation dust generation. Even commercial products have developed equipment that does not require anchor installation work, but it can only be applied to flat walls and cannot be applied to curved walls such as bioshields. To solve this problem, a method of fixing to the scaffolding pipe was designed. The equipment developed based on this method fundamentally blocks the generation of radioactive dust. The vertical position can be adjusted using guide shafts and jack screws, and the horizontal position can be adjusted using scaffolding clamps. In addition, the distance between the installation location and the wall can be adjusted by adjusting the scaffolding clamp location of the device. Lastly, it can be rotated to the left and right, so that even on a curved wall, the sampling position can be performed perpendicular to the wall.

Core drills that take specimens for measuring compressive strength use the wet type. Core drilling by wet type in radioactively contaminated concrete leads to the disposal of sludge as radioactive waste. Water supplied during core drilling is scattered in all directions by the rotation of the core drill bit, which causes radiation exposure to workers, so measures must be taken to ensure that the water does not splash and gather in one place. Nileplant Co., Ltd. has developed a sludge collection device that can be used with a core drilling device. It can be inserted into a 4-inch core drill bit to meet the specimen regulations of ASTM C 42/C 42M, and nylon resin was used as a material to withstand friction with water, and the wall of the drainage part was thickened to increase durability.

Based on these results, it is expected to be able to work more quickly and safely when collecting core drilling samples of radioactively contaminated concrete or radiation and concrete.

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Keywords: Concrete core drilling, Radiologically contaminated concrete, Concrete degradation, Concrete activation, Core drilling sludge treatment

A Study to Develop a Scenario for Cutting Large Structures During Nuclear Power Plant Decommissioning

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Among the twenty six nuclear power plants in Korea, twenty four plants are currently in operation excluding the two permanently shut-down Kori #1 and Wolsung #1 plants. The decommissioning process includes many tasks such as cutting, decontamination, disposal and treatment. Among the tasks, because cutting is one of the tasks performed close to the target structure, there is a possibility for the workers to be exposed excessively to the radiation.

There are representative large structures such as steam generators, nuclear reactors, reactor coolant pump, and pressurizer, made of metals, and radioactive concrete, made of concrete. Especially, compared to the trend of research to manage the radiation exposure of steam generators that are directly connected to pressurizers, the trend of research to manage the radiation exposure of pressurizers to workers is not satisfactory. Moreover, although there have been many studies on radioactive concrete, the studies to manage the radiation exposure to workers with a systematic cutting scenario are insufficient.

In this study, radioactive concrete, a representative large structure made of concrete, was selected as the target for evaluation. The conditions for evaluation were cutting speed ($1\sim 10\text{ m}^2/\text{hr}$) and the time for cutting (permanent shutdown ~ 30 years after the shutdown). A cutting scenario was developed by applying the situation for abrasive decontamination beforehand and Hot-to-Cold and Cold-to-Hot, and effort was made to derive a reasonable plan.

The evaluation result derived were hourly radiation dose distribution of $1.19\sim 0.103\text{ mSv/hour}$ and $1.29\sim 0.0113\text{ mSv/hour}$ for a scenario without abrasive decontamination (in the order of Hot to Cold, Cold to Hot), and hourly radiation dose distribution of $0.547\sim 0.0479\text{ mSv/hour}$ and $0.608\sim 0.0522\text{ mSv/hour}$ for a scenario with abrasive decontamination. The maximum value of collective dose derived was $1.54\text{E}+04\text{ mSv}$ at the cutting time of permanent shutdown with cutting speed of $1\text{ m}^2/\text{hour}$ in the Cold to Hot scenario before abrasive decontamination, and the minimum value derived was $5.15\text{E}+01\text{ mSv}$ at the cutting time of 30-year after the permanent shutdown with cutting speed of $10\text{ m}^2/\text{hour}$ in the Hot to Cold scenario after abrasive decontamination.

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Keywords: Decommissioning, Cutting scenario, Exposure dose, Bio-shield

Activation Assessment and Waste Classification of End Shield in Wolsong Unit 1

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Wolsong unit 1, the first PHWR (Pressurized Heavy Water Reactor) in Korea, was permanent shut down in 2019. In Korea, according to the Nuclear Safety Act, the FDP (Final Decommissioning Plan) must be submitted within 5 years of permanent shutdown. According to NSSC Notice, the types, volumes, and radioactivity of solid radioactive wastes should be included in FDP chapter 9, Radioactive Waste Management. Therefore, in this study, activation assessment and waste classification of the End shield, which is a major activation component, were conducted.

MCNP and ORIGEN-S computer codes were used for the activation assessment of the End shield. Radioactive waste levels were classified according to the cooling period of 0 to 20 years in consideration of the actual start of decommissioning. The End shield consists of Lattice tube, Shielding ball, Sleeve insert, Calandria tube shielding sleeve, and Embedment Ring. Among the components composed for each fuel channel, the neutron flux was calculated for the components whose level was not predicted by preliminary activation assessment, by dividing them into three channel regions: central channel, inter channel, and outer channel. In the case of the shielding ball, the neutron flux was calculated in the area up to 10 cm close to the core and other parts to check the decrease in neutron flux with the distance from the core.

The neutron flux calculations showed that the highest neutron flux was calculated at the Sleeve insert, the component closest to the fuel channel. It was found that the neutron flux decreased by about 1/10 to 1/20 as the distance from the core increased by 20 cm. The outer channel was found to have about 30% of the neutron flux of the center channel. It was found that no change in radioactive waste level due to decay occurred during the 0 to 20 years cooling period.

In this study, activation assessment and waste classification of End Shield in Wolsong unit 1 was conducted. The results of this study can be used as a basis for the preparation of the FDP for the Wolsong unit 1.

Keywords: Activation, Waste classification, Wolsong unit 1, Decommissioning

Fabrication of Membrane Electrode Assemblies for PEM Electrolysis of Tritiated Water

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For decontamination and quantification of trace amount of tritium in water, an efficient separation technology capable of enriching tritium in water is required. Electrolysis is a key technology for tritium enrichment as it has a high H/T and D/T separation factors. To separate tritium, it is important to develop a proton exchange membrane (PEM) electrolyzer having high hydrogen isotope separation factor as well as high electrolyzer cell efficiency. However, there has not been sufficient research on the separation factor and cell efficiency according to the composition and manufacturing method of the membrane electrode assembly (MEA). Therefore, it is necessary to study the optimal composition and manufacturing method of the MEA in PEM electrolyzer.

In this study, the H/D separation factor and water electrolysis cell efficiency of PEM electrolyzer were analyzed by changing the anode and cathode materials and electrode deposition method of the MEA. After the water electrolysis experiment using deionized water, the D/H ratio in water and hydrogen gas was measured using a cavity ring down spectrometer and a mass spectrometer, respectively, and the separation factor was calculated. To calculate the cell efficiency of water electrolysis, a polarization curves were obtained by measuring the voltage changes while increasing the current density.

As a result of the study, the water electrolyzer cell efficiency of the MEA fabricated with different anode/cathode configurations and electrode formation methods was higher than that of commercial MEA. On the other hand, the difference in H/D separation factor was not significant depending on the MEA fabrication methods. Therefore, using a cell with high cell efficiency when the separation factor is the same will help construct a more efficient water electrolysis system by lowering the voltage required for water electrolysis.

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Keywords: Hydrogen isotope, Separation, PEM (proton exchange membrane), Water electrolysis

A Study on the Site Selection Method of Radioactive Waste Treatment Facility for the Decommissioning of Nuclear Facilities

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The decommissioning of Korea's nuclear power facilities is expected to take place starting with the Kori Unit 1 followed by the Wolsong Unit 1. In Korea, since there is no experience of decommissioning, considerations of site selection for the waste treatment facilities and reasonable selection methods will be needed. Only when factors to be considered for construction are properly selected and their effects are properly analyzed, it will be possible to operate a treatment facility suitable for future decommissioning projects. Therefore, this study aims to derive factors to be considered for the site selection of treatment facilities and present a reasonable selection methodology through evaluation of these factors.

In order to select a site for waste treatment facilities, three virtual locations were applied in this study: warehouse 1 to warehouse 3. Such a virtual warehouse could be regarded as a site for construction warehouses, material warehouses, annexed building sites, and parking lots in nuclear facilities.

If the selection of preliminary sites was made in the draft, then it is necessary to select the influencing factors for these sites. The site of the treatment facility shall be suitable for the transfer of the waste from the place where the dismantling waste is generated to the treatment facility. In addition, in order for construction to take place, interference with existing facilities and safety should not be affected, and it should not be complicated or narrow during construction. Considering the foundation and accessibility, the construction of the facility should be economical, and the final dismantling of the facility should also be easy.

In order to determine one final preferred plan with three hypothetical locations and five influencing factors, there will be complex aspects and it will be difficult to maintain consistency as the evaluation between each factor progresses. Therefore, we introduce the Analytic Hierarchical Process (AHP) methodology to perform pairwise comparison between factors to derive an optimal plan.

One optimal plan was selected by evaluating the three virtual places and five factors of consideration presented in this study. Given the complexity and consistency of multiple influencing factors present and prioritizing them, AHP tools help users make decisions easier by providing simple and useful features. Above all, it will be most important to secure sufficient grounds for pairwise comparison between influencing factors and conduct an evaluation based on this.

Keywords: Decommissioning, AHP, Pairwise comparison, Decision making tool

Aspects to be Considered in the Development of the Decommissioning Process of Nuclear Power Plants

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The domestic Nuclear Power Plant (NPP) decommissioning project is expected to be carried out sequentially, starting with Kori Unit 1. As a license holder, in order to smoothly operate a new decommissioning project, a process in terms of project management must be well established. Therefore, this study will discuss what factors should be considered in establishing the process of decommissioning NPPs.

Various standards have been proposed as project management tools on how to express the business process in writing and in what aspects to describe it. Representatively, PMBOK, ISO 21500, and PRICE 2 may be considered.

It will be necessary to consider IAEA safety standards in the nuclear decommissioning project. GSR part 6 and part 2 can be considered as two major requirements. GSR part 6 presents a total of 15 requirements, including decommissioning plans, general safety requirements until execution and termination. GSR part 2 presents basic principles for securing the safety of nuclear facilities, and there are a total of 14 requirements.

Domestic regulatory guidelines should be considered, and there will be largely laws and regulations related to the decommissioning of nuclear facilities, guidelines for regulatory agencies, and guidelines and regulations related to HSE. The Nuclear Safety Act, Enforcement Decree, Enforcement Rules, and NSSC should be considered in the applicable law for nuclear facilities.

Since the construction and operation process has been established for domestic decommissioning project, there will be parts where existing procedures must be applied in terms of life cycle management of facilities and the same performance entity. As a management areas classification in the construction and operation stage, it seems that a classification similar to Level 1 and Level 2 should be applied to the decommissioning project.

This study analyzed the factors to be considered in the management system in preparing for the first decommissioning project in Korea. Since it is project management, it is necessary to establish a system by referring to international standards, and it is suggested that domestic regulatory reflection, existing business procedures, and domestic business conditions should be considered.

Keywords: Decommissioning, PMBOK, Project management, Process

Considerations on the Linkage Between Decommissioning Cost Items and Items in the ISDC

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In the case of decommissioning nuclear facilities in Korea, the dismantling activities will be initiated after obtaining approval from the regulatory agency for the Final Decommissioning Plan (FDP). The contents to be described in the FDP are presented in the notice of the Nuclear Safety and Security Commission, and among them, it is suggested to estimate and provide the basis for the decommissioning cost. The Work Breakdown Structure (WBS) is used for schedule management for the project, and the performance activities can be used as a cost management structure as well as schedule management. In order to easily manage the process and cost, the WBS structure can be normally used, and at this time, there might be a connection with ISDC if necessary. Therefore, this study aims to examine the link between activities from the WBS structure to ISDC in the decommissioning project.

In general, the activities assumed as a WBS structure in this study in carrying out the decommissioning project were derived at the Level 1 and Level 2. Activities at Level 1 can be classified into project management costs, controlled area dismantling, conventional area dismantling, site remediation, waste treatment facility, construction/service, R&D, waste treatment and disposal, and characterization. For Level 2 activities, a cost activities embodied in Level 1 was derived.

ISDC was developed by the OECD Nuclear Energy Agency (NEA) decommissioning cost estimation group, which improves ambiguous cost systems and presents common cost items for direct comparison between international decommissioning projects. The ISDC consists of Level 1, Level 2, and Level 3, where Level 1 represents the principal activity, Level 2 represents the activity group, and Level 3 represents the typical activity. The cost categories for typical activity at Level 3 consist of labour, investment, expenses, and contingency.

In this study, the connection between WBS and ISDC was shown, and a comparison was made at Level 2. Directly, one-on-one matches have difficulties, and as much as possible, they were organized into similar items. We arranged the Level 2 linked to ISDC based on WBS. If there is a difficulty in one-on-one matches, it seems that the accurate cost calculation of ISDC items should consider the impact of additional cost distribution. Therefore, in order to calculate ISDC costs, it seems necessary to organize cost items of WBS in consideration of the ISDC.

Keywords: Decommissioning, Cost estimation, ISDC, Decommissioning activities

Proposal for the Document Types to be Managed During the Decommissioning of Nuclear Power Plants

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The nuclear power plant decommissioning project inevitably considers time, cost, safety, document, etc. as major management areas according to the PMBOK technique. Among them, document management, like all projects, will be an area that must be systematically managed for the purpose of information delivery and record maintenance. In Korea, where there is no experience in the decommissioning project yet, data management is systematically managed and maintained during construction and operation. However, if the decommissioning project is to be launched soon, it is necessary to prepare in consideration of the system in operation, what difference will occur from it in terms of data management, and how it should be managed.

As a document that can occur in the decommissioning project, this study was considered from the perspective of the licensee. Therefore, the types of documents that can be considered at Level 1 can be divided into (1) corresponding documents, (2) project documents, (3) internal documents, and (4) reference materials. Four document types are recommended based on Level 1 for the classification of documents to be managed in the decommissioning of nuclear facilities.

In this study, documents to be managed in the decommissioning project of nuclear facilities were reviewed and the type was to be derived. Although it was preliminary, it was largely classified into major categories 1, middle categories 2, and 3 levels, and documents that could occur in each field were proposed. As a result, it could be largely classified into corresponding documents, project documents, internal documents, and reference materials, and subsequent classifications could be derived. Documents that may occur in the decommissioning project must be managed by distinguishing between types to reduce the time for duplication or search, and the capacity of the storage can be efficiently managed. Therefore, it is hoped that the document types considered in this study will be used as reference materials for the decommissioning project and develop into a more systematic structure.

Keywords: Decommissioning, Document type, Project management

A Review on Liquid Radioactive Waste Processing System for Radioactive Waste Treatment Facility

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The type of radioactive waste that may occur in the process of NPP dismantling can be classified into solid, liquid, gas, and mixed waste. Most of the radioactive waste generated during the dismantling of a NPP is metal solid waste, but liquid radioactive waste is also a very important factor in terms of radiation environmental impact assessment.

In the case of liquid radioactive waste, it is necessary to calculate the generation amount in order to design liquid radioactive waste processing system of Radioactive Waste Treatment Facility (RWTF). Depending on the amount of liquid radioactive waste generated, the type of liquid radioactive waste processing system included in the RWTF is different. In addition, in order to apply to the domestic RWTF, it is important to secure the site area occupied by the each system, the liquid radioactive waste treatment capacity of the system, and how to secure circulating water used for dilution and discharge of liquid radioactive waste.

Therefore, this review aims to suggest an optimal method for the treatment system for liquid radioactive waste included in RWTF of Wolsong.

Keywords: Decommissioning, Liquid radioactive waste, RWTF, Radioactive waste

A Review on Liquid Radioactive Waste Generated for Decommissioning of NPP

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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 liquid radioactive waste, it is necessary to calculate the generation amount in order to treat radioactive waste at a Radioactive Waste Treatment Facility (RWTF) or on-site.

In this regard, there is no Code and Standard for the amount of liquid radioactive waste generated during NPP are dismantled, but ANSI/NS-55.6 describes the amount of liquid radioactive waste generated from a light water reactor type NPP. This code is applied to nuclear power-related facilities such as domestic NPP and radioactive waste disposal facility.

Therefore, this review intends to suggest an application plan for domestic NPP decommissioning through codes for liquid radioactive waste expected to generate during nuclear power plant decommissioning.

Keywords: Decommissioning, Liquid radioactive waste, Decontamination, Radioactive waste

Crystallized Glass Tailored by Controlled Heat Treatment for Carbon Dioxide Capture Under Mild Conditions

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Radioactive carbon dioxide ($^{14}\text{CO}_2$) capture using innovative materials is desirable due to associated radiological hazards, and growing climate change. Mineral carbonation technology (MCT) is amenable to irreversibly capture CO_2 . Typically, MCT is attractive because capturing carbon through the chemical reaction between alkaline earth metal ions and CO_2 forms insoluble and significantly stable carbonates. However, most applications of MCT have an intrinsic restriction regarding their operational conditions since no forward reaction occurs within realistic time scales. Thereby, the CO_2 capture performance, such as CO_2 capacity and carbonation reaction rate, of MCTs and their applications are severely restricted by the difficulty of operations under mild conditions. For example, natural minerals require aggressive carbonation reaction conditions e.g. high pressure (≥ 20 bar), high temperature (> 373 K), and pH-adjusted carrier solutions. To overcome such obstacles, the fabrication of alkaline earth oxides impregnated into an amorphous glass structure have been recently developed. They show enhanced rates of dissolution of alkaline earth metal ions and carbonation reaction due to the loosely packed glass structure and the generation of a surface coating silica gel, consequently facilitating CO_2 capture under mild conditions.

In this presentation, we report the synthesis and application of a crystallized glass tailored by controlled heat treatment for CO_2 capture under mild conditions. The controlled heat treatment of an alkaline earth oxide-containing glass gives rise to a structural transformation from amorphous to crystalline. The structural characterizations and CO_2 capture performance, including CO_2 capacity, carbonation reaction rate, and the dissolution rate of alkaline earth metal ion, were analyzed to reveal the impact of controlled heat treatment and phase transformation.

Acknowledgements

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Keywords: Glass adsorbent, CO_2 capture, Carbon 14, Radioactive waste, Mineral carbonation

Solidification of Radioactively Contaminated Soils Using Cement Waste Forms

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The soils contaminated with radionuclides such as Cs-137 and Sr-90 should be solidified using a binder matrix, because radioactively contaminated soils pose environmental concerns and human health problems. Ordinary Portland cement has been widely used to solidify various radioactive wastes due to its low cost and simple process. In this study, simulant soil waste was solidified using cement waste form. The soils were collected around 'Kori Nuclear Power Plant Unit 1' and they were contaminated with the prepared simulant liquid waste containing Fe, Cr, Cs, Ni, Co, and Mn. The water-to-dry ingredients (W/D) ratio of cement waste form was 0.40. The cement paste was poured into a cubic mold (5×5×5 cm) and then cured for 28 days at room temperature. The 28-day compressive strength, water immersion, and EPA1311-toxicity characteristic leaching procedure (TCLP) tests were performed to evaluate the structural stability of cement waste form. The compressive strength was not proportional to soil waste loading, and the lowest compressive strength (4±0.1 MPa) was achieved in cement waste form containing 50wt% soil waste. After the water immersion test for 90 days, the compressive strength of cement waste form with 50wt% soil waste increased to 7.5±0.6 MPa, meeting the waste form acceptance criteria in the repository. It is believed that long-term water immersion test contributed to the additional curing and hydration reaction, resulting in the enhanced compressive strength. As a result of the TCLP test, the released amount of As, Ba, Cd, Cr, Pb, Se, Co, Cs, and Sr was less than the domestic and international standards. These results imply that cement waste form can be a promising candidate for the solidification of radioactive soil wastes.

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Keywords: Radioactive contaminated soil, Cement immobilization, Compressive strength, Immersion test, TCLP

Introduction to Research and Development Tasks in Remediation of Radioactively Contaminated Sites in Permanent Shutdown Nuclear Power Plant

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Kori-1 and Wolsong-1 nuclear power plants were permanently shut down in June 2017 and December 2019, and are currently in the preparation stage for decommissioning. In this regard, it is necessary to secure nuclear power plant decommissioning capacity in preparation for the domestic decommissioning marketplace. To address this, the Korea Research Institute of Decommissioning (KRID) was established to build a framework for the development of integrated nuclear decommissioning technology to support the nuclear decommissioning industry. The institute is currently under construction in the Busan-Ulsan border area, and a branch is planned to be established in the Gyeongju area.

Recently, R&D projects have been launched to develop equipment for the demonstration and support verification of decommissioning technology. As part of the R&D project titled “Development and demonstration of the system for radioactivity measurement at the decommissioning site of a nuclear power plant”, we introduce the plan to develop a radioactivity measurement system at the decommissioning site and establish a demonstration system. The tasks include (1) measurement of soil radioactive contamination and classification system, (2) visualization system for massive dismantling of nuclear facilities, (3) automatic remote measurement equipment for surface contamination, and (4) bulk clearance verification equipment. The final goal is to develop a real-time measurement and classification system for contaminated soil at the decommissioning site, and to establish a demonstration system for nuclear power plant decommissioning.

The KRID aims to contribute and support the technological independence and commercialization for domestic decommissioning sites remediation of nuclear power plant decommissioning site by establishing a field applicability evaluation system for the environmental remediation technology and equipment demonstration.

Keywords: KRID, Decommissioning, Remediation, Measurement equipment, Demonstration, Verification

Introduction of Radionuclides in Feeder Piping During CANDU Heat Transport System Decontamination

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Canada's Pickering Unit 3 was performed a three-stage decontamination from June to August 1989 in preparation for pressure tube replacement. The first step was a reducing CAN-DECON treatment to dissolve the magnetic film inside the reactor, which was applied following partial defueling of the reactor core. The second step was an oxidative dilute alkaline permanganate treatment to remove the chromium-rich oxides of the stainless steel parts. And the final CAN-DECON step was applied continuously after completely removing fuel from the reactor core. In situ pipe gamma-ray spectroscopy techniques were applied to measure radioactivity within feeder piping during various stages of Pickering Unit 3 decontamination. Measurements were performed at a maximum dose rate of 5 mSv/h, and both the detector and the scanned feeder pipe were properly shielded from other neighboring pipes. ^{60}Co was the dominant radionuclide in feeder piping prior to decontamination. And radionuclides ^{103}Ru , ^{95}Zr , ^{95}Nb , ^{59}Fe , ^{140}La and ^{124}Sb were detected. The Co-60 radioactivity was $2.09 \times 10^5 \text{ Bq/cm}^2$ before decontamination and $3.11 \times 10^3 \text{ Bq/cm}^2$ after decontamination in the inlet feeder pipe T18. And in the outlet feeder pipe P21, it is $2.56 \times 10^4 \text{ Bq/cm}^2$ before decontamination and $2.04 \times 10^3 \text{ Bq/cm}^2$ after decontamination.

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Keywords: CANDU, Heat transport system, Gamma-ray spectrometry, Decontamination, ^{60}Co radioactivity

Vitrification of Dry Wastes From Nuclear Power Plants in China

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With the rapid growth of nuclear power in China, a large number of dry wastes, which mainly include the high efficiency particulate air filters (glass fiber), cotton, polyethylene, and absorbent paper with low-level radioactivity and high volume, will be produced during the operation and maintenance of the nuclear power plants. Thermal plasma treatment is a world acceptable technology to incinerate and immobilize radioactive wastes, owing to the high volume reduction factor and the excellent chemical durability of the vitrified waste form. China has developed thermal plasma technology for the treatment of dry wastes from nuclear power plants for more than 15 years and the pilot plant has been constructed. This work will concentrate on the formulation of waste glass fiber to adapt to the vitrification process. A three-component (glass fiber-CaO-Na₂O) constrained-region mixture experiment was designed and their viscosity data was mainly studied. The quadratic Scheffé model was used to plot the component effect on melting temperature. The retentions of simulated nuclides, such as Co, Sr, and Cs in the glasses were analyzed. In addition, the glass fiber as a glass matrix to immobilize residual ashes from the thermal plasma gasification of cotton, polyethylene, and absorbent paper was investigated as well.

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Keywords: Dry waste, Thermal plasma treatment, Vitrification, Volatilization

Simulation-Based Alternative Evaluation to Optimize RWTF for Decommissioning of NPP

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When decommissioning a nuclear power plant, a large amount of radioactive waste is generated simultaneously. Therefore, efficient treatment of radioactive waste is crucial to the success of the decommissioning process. An utility or decommissioning contractor of NPP often build separate radioactive waste treatment facilities (RWTF) to handle this waste. In Korea, RWTFs are planned to be built for the decommissioning of the Kori Unit 1 and Wolsong Unit 1.

In this study, we introduce an application case of using process simulation to derive the optimal layout design and investment plan for a radioactive waste treatment facility. In particular, the steam generator is the largest and most complex device processed in RWTF. Therefore, it is necessary to reflect the large equipment processing area that can treat steam generators in the design of RWTF.

In this study, Siemens' Plant Simulation[®] was used to derive an optimization plan for the dismantling area of large equipment in RWTF.

First, a virtual facility was built by modeling based on the steam generator dismantling process and facilities developed by Doosan Enerbility. This was used to pre-validate the facility investment plan, discover wasteful factors in the logistics waste streams, and evaluate alternatives to derive, validate, and apply appropriate improvement alternatives. Through this, we designed a layout based on the optimal logistics waste streams, appropriate workstations, and the number of buffer places. In addition, we propose various optimization measures such as investment optimization based on optimal operation of facility resources such as facilities and manpower, and establishment of work standards.

Keywords: Dismantling, Reactor pressure vessel, Mechanical cutting, Mockup

Application Evaluation of Plasma Milling Robot for SUS 316 L and Alloy 600

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Various cutting technologies such as thermal and mechanical are being researched and developed to dismantle shutdown nuclear power plants. Each technology has the following advantages and disadvantages. The thermal cutting method has low reaction force and fast cutting speed, but secondary waste such as fume, dross, and fine dust is generated. The mechanical cutting method has the advantage of low generation of secondary waste such as fume, dross, and fine dust, but has the disadvantage of increasing the size of the device due to its large reaction force. In this study, the performance of plasma milling robot cutting technology for nuclear power plant materials was evaluated. First, before applying plasma auxiliary milling to the robot, tests were conducted on SUS 316 L and Alloy 600 to secure processing conditions such as plasma torch output and transfer speed. The test have shown that the mechanical strength was decreased of each material at the output power of the plasma torch of 4.4 and 8.4 kW, the transfer speed of 200 and 100 mm/min. Based on the test results, a plasma milling was attached to the robot and tested, and it was confirmed that even a small robot with a load of 140 kg can cut without any major problems.

Keywords: Plasma, Transferred type torch, Hybrid cutting, SUS 316 L, Alloy 600

Development of Gamma-ray Visualization Detector for NPP Decommission

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Prevention of radiation hazards to workers and the environment in the event of decommissioning nuclear power plants is a top priority. To this end, it is essential to continuously perform radiation characterization before and during decommissioning.

In operating nuclear power plants, various detectors are used depending on the purpose of measurement. Portable detectors used in power plants have excellent portability, but there is a limit to the use of a single measuring device alone to quantify radioactive contamination, nuclide analysis, and ensure representation of measurement results.

In foreign countries, gamma-ray visualization detectors are being actively used for operating and decommissioning nuclear power plants. KHNP is also conducting research on the development of gamma-ray visualization detectors for multipurpose field measurement at decommissioning nuclear power plants. It aims to develop detectors capable of visualizing radioactive contamination, analyzing nuclides, estimating radioactivity, and estimating dose rates. To this end, we are developing related software according to the development process by purchasing sensors from H3D, which account for more than 75% of the US gamma-ray visualization detector market. In addition, field tests are planned in the order of Wolsong Unit 1 and Kori Unit 1 with Research reactor in Gongneung-dong in accordance with the progress of development. The detector will be optimized by analyzing the test results according to various gamma radiation field environments.

The development detector will be used for various measurement purposes for Kori unit 1 and Wolsong unit 1.

Keywords: Gamma-ray visualization detector, Measurement, Decommissioning

Deriving Factors and Solutions Affecting Resident Acceptance in Land Transport of Radioactive Waste

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A large amount of radioactive wastes are generated during the decommissioning of the nuclear power plant. The need for radioactive waste management is emerging as the permanent suspension of domestic nuclear power plants Gori Unit 1 and Wolsong Unit 1 has been decided. According to the analysis of the number of land transportation of Gori Unit 1, when the dismantling period is expected to be five years, the number of land transportation is 459. Accordingly, measures are needed to improve the acceptability of residents on land transportation routes. Currently, there is little preparation for the acceptance of residents by transporting radioactive waste in Korea. This study analyzed the literature related to radioactive waste and nuclear power generation to derive factors that affect the acceptance of residents on the transport route of radioactive waste.

Factors Affecting Resident Acceptability

- Mistrust of the measured dose levels themselves
- fear of radiation
- Lack of training in radiation, nuclear power
- Insufficient information for easy identification of dose and concentration
- economic compensation

In order to improve the acceptability of residents when transporting radioactive waste on land, it is necessary to scientifically prove the safety of the low and intermediate level radioactive waste transport path, and policy improvement considering the acceptability of residents is needed. In subsequent studies, it is necessary to specifically derive solutions to the above factors. This study is significant in that it derived factors that could affect low and intermediate level radioactive waste transportation, considering that no countermeasures have been prepared to improve the acceptance of residents by transporting low and intermediate level radioactive waste in Korea.

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Keywords: NPP decommissioning, Radioactive waste, Transportation, Resident acceptance

Dose Assessment Method for Heavy Water Reactor Decommission Workers

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KHNP is carrying out international technical cooperation and joint research projects to decommission Wolsong unit 1 reactor. Construction data of the reactor structures, experience data on the pressure tube replacement projects, and the operation history were reviewed, and the amount of dismantled waste was calculated and waste was classified through activation analysis.

By reviewing COG (CANDU owners Group) technical cooperation and experience in refurbishment projects, KHNP's unique Wolsong unit 1 reactor decommissioning process was established, and basic design of a number of decommissioning equipment was carried out. Based on this, a study is being conducted to estimate the worker dose of dismantling workers.

In order to evaluate the dose of external exposure of dismantling workers, detailed preparation and dismantling processes and radiation field evaluation of activated structures are required. The preparation process can be divided into dismantlement of existing facilities that interfere with the reactor dismantling work and construction of various facilities for the dismantlement process. Through process details, the work time, manpower, and location required for each process will be calculated.

Radiation field evaluation takes into account changes in the shape of structures by process and calculates millions of areas by process, so integrated scripts are developed and utilized to integrate input text data. If the radiation field evaluation confirms that the radiation risk of workers is high, mutual feedback will be exchanged so that the process can be improved, such as the installation of temporary shields.

The results of this study will be used as basic data for the final decommissioning plan for Wolsong unit 1. By reasonably estimating the dose of workers through computer analysis, safety will be the top priority when decommissioning.

Keywords: COG, Technical cooperation, Wolsong unit 1, Decommissioning

International Technical Cooperation on Decommissioning of Heavy Water Reactor

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According to IAEA PRIS, there is no record of dismantling commercial heavy water reactors among 57 heavy water reactors around the world. In Canada, which has the largest number of heavy water reactors, three of the 22 commercial heavy water reactors with more than 500 MW are permanently suspended, Gentilly unit 2 (2012), Pickering unit 2 (2007), and Pickering unit 3 (2008), all of which chose a delayed decommissioning strategy. On the other hand, Wolsong unit 1, which will be the world's first heavy water reactor to be dismantled commercially, will be immediately carried out as a decommissioning strategy.

KHNP has established various cooperation systems with advanced companies and international organizations related to overseas NPP decommission and is actively exchanging technologies. Among them, the most important focus is on research cooperation related to COG (CANDU owners Group). The first case is a joint study on Conceptual Calandria Segmentation. Four areas of process, waste management, ALARA, and cost for decommissioning reactors to be submitted to Canadian regulators for approval of Pickering and Gentilly-2's preliminary decommissioning plan have been evaluated, and research on Wolsong unit 1 is currently underway. The second case is Decommissioning and long-term waste management R&D. Although the technical maturity is low, it studies the common interests of member companies in the decommissioning of heavy water reactor power generation companies and long-term waste management. Robotics for dismantling high-radiation structures, C-14, H-3 measurement and removal methods, and concrete decontamination technology, which are characterized by heavy water, are being actively studied.

KHNP is strengthening international cooperation with COG to prepare for the successful decommissioning of Wolsong unit 1. Based on previous studies by Pickering and Gentilly-2, an evaluation of the decommissioning of Wolsong unit 1 reactor is being conducted. In addition, it is preparing for decommissioning through experience analysis of the pressure tube replacement project.

Keywords: COG, Wolsong unit 1, Decommissioning

Development of Real-time Worker Safety Evaluation Technology Based on the Measurement of Dose Distribution and Worker Location for Decommissioning

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Decommissioning plan of nuclear facilities require the radiological characterizations and the establishment of a decommissioning process that can ensure the safety and efficiency of the decommissioning workers. By utilizing the rapidly developed ICT technology, we have developed a technology that can acquire, analyze, and deliver information from the decommissioning work area to ensure the safety of decommissioning workers, optimize the decommissioning process, and actively respond to various decommissioning situations. The established a surveillance system that monitors nuclide inventory and radiation dose distribution at dismantling work area in real time and wireless transmits data for evaluation. Developed an evaluation program based on an evaluation model for optimizing the dismantling process by linking real-time measurement information. We developed a technology that can detect the location of dismantling workers in real time using stereovision cameras and artificial intelligence technology.

The developed technology can be used for safety evaluation of dismantling workers and process optimization evaluation by linking the radionuclides inventory and dose distribution in dismantling work space of decommissioning nuclear power plant in the future.

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Keywords: Decommissioning, Worker safety, Optimization, Characterization

Development of Laser Ablation Technique and Transfer Robot for Removal of a Radioactively Contaminated Inside Oxide Layer Coolant Pipes in Nuclear Power Plants

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Radioactive Oxide is formed on the surface of the coolant pipe of the nuclear power plant. In order to remove the oxide film that is formed on the surfaces of the coolant pipe, chemical and physical decontamination technologies are used. The disadvantage of traditional technologies is that they produce secondary radioactive wastes. Therefore, in this study, the short-pulsed laser eco-friendly technology was used in order to reduce the production of secondary radioactive wastes. It was also used to minimize the damage that was caused to the base material and to remove the contaminated oxide film. The study was carried out using a Stainless steel 304 specimen that was coated with nickel-ferrite particles. Additionally, a transport robot was 3D modeled and manufactured in order to efficiently remove the oxide film from the coolant pipe of the nuclear power plant. The transport robot has a fixed laser head to move inside the horizontal and vertical pipes. The rotating laser head removes the contaminated oxide film on the inner surface of the coolant pipe. In the future, as a condition of the 1064nm short-pulsed laser ablation technique determined by basic analysis, we plan to analyze whether the transport robot is applicable to the radiation contamination site of the nuclear power plant.

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Keywords: Primary system, Decontamination technology, Radioactive secondary waste, Oxide film, Laser ablation

Study on Uranium Adsorption Using Phosphate-Functionalized Silica and Experimental Optimization: Addressing Initial Concentration and pH as Key Factors

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In this study, we evaluated the performance of phosphate-functionalized silica in adsorbing uranium and provided insights into optimizing the initial conditions of the uranium solution (concentration and pH), which are often overlooked in uranium adsorption studies. While most studies take into account the effect of pH on both the surface charge of the adsorbents and the dissolved speciation of uranium in solution, they often overlook the formation of solid phases such as $\beta\text{-UO}_2(\text{OH})_2$ (cr) and $\text{UO}_3 \cdot 2\text{H}_2\text{O}$ (cr), leading to an overestimation of the adsorption capacity. To address this issue, we considered the speciation of U(VI) calculated using thermodynamic data. Our findings suggest that it is reasonable to evaluate the adsorption performance at pH 4 and concentration below 1.35 mM. The formation of $\beta\text{-UO}_2(\text{OH})_2$ (cr) starts at 23 μM (pH 5) and 1 μM (pH 6) and increases sharply with increasing concentration. To avoid interference from the formation of solid phases, experiments should be conducted at lower concentrations, which in turn require very small $m_{\text{sorbent}}/V_{\text{solution}}$ ratios. However, controlling small amounts of sorbent can be challenging, and increasing the volume of the solution can generate significant amounts of radioactive waste. We also used UV-vis spectra analysis to investigate the formation of solid phases. We found that a 100 mg L⁻¹ uranium solution resulted in the formation of colloidal particles in the solid phase after 2.5 hours at pH 6, while at pH 4, no significant changes in absorbance were observed over 120 hours, indicating a stable ion phase. Based on these conditions, we obtained an excellent adsorption capacity of 110 mg g⁻¹.

Keywords: Silica, Uranium adsorption, Phosphate-functionalization, pH, Concentration

Electrodeposition of Aluminum Coatings on Copper Substrates Using Deep Eutectic Solvents (DES) in the Air

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Aluminum's exceptional properties, such as its high strength-to-weight ratio, excellent thermal conductivity, corrosion resistance, and low neutron absorption cross-section, make it an ideal material for diverse nuclear industry applications, including aluminum plating for the building envelope of nuclear power plants. However, plating aluminum presents challenges due to its high reactivity with oxygen and moisture, thus, complicating the process in the absence of controlled environments. Plating under an inert atmosphere is often used as an alternative. However, maintaining an inert atmosphere can be expensive and presents an economic challenge. To address these challenges, an innovative approach is introduced by using deep eutectic solvents (DES) as a substitute for traditional aqueous electrolytes due to the high solubility of metal salts, and wide electrochemical window. In addition, DESs offer the benefits of low toxicity, low flammability, and environmentally friendly, which makes DESs candidates for industrial-scale applications. In this study, we employed an AlCl_3 -Urea DES as the electrolyte and investigated its potential for producing aluminum coatings on copper substrates under controlled conditions, for example, current density, deposition duration, and temperature. A decane protective layer, non-polar molecular, has been used to shield the AlCl_3 -Urea electrolyte from the air during the electrodeposition process. The electrodeposition was successful after being left in the air for two weeks. This study presents a promising and innovative approach to optimizing aluminum electrodeposition using deep eutectic solvents, with potential applications in various areas of the nuclear industry, including fuel cladding, waste encapsulation, and radiation shielding.

Acknowledgements

This research was supported by the Fundamental Research Program of the Korean Institute of Materials Science (Grant PNK8430).

Keywords: Aluminum, Electrodeposition, Deep eutectic solvents, Protective layer, Open atmosphere

A Study on the Evaluation of Worker Exposure Dose According to the Derivation of Work Analysis Amount for Each Single/Continuous Process

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Radioactive waste generated during nuclear power plant decommissioning is classified as radioactive waste before the concentration is identified, but more than 90% of the amount generated is at a level that can be by clearance. However, due to a problem in the analysis procedure, the analysis is not carried out at the place of on-site but is transported to an external institution to identify concentration, which implies a problem of human error because 100% manual. As a way to solve this problem, research is underway to develop a mobile radioactive waste nuclide analysis facility.

The mobile radionuclide analysis facility consists of a preparation room, a sample storage room, a measurement room, a pretreatment room, and a waste storage room, and is connected to an external ventilation facility. In addition, since the automation module is built-in from the sample pre-threatening step to the separation step, safety can be improved and rapid analysis can be performed by being located in the decommissioning site. As an initial study for the introduction of a mobile nuclide analysis facility, Visiplan, a preliminary external exposure evaluation code, was used to derive the analysis workload by a single process and evaluate the exposure dose of workers. Based on this, as a follow-up study, the amount of analysis work according to the continuous process and the exposure dose of workers were evaluated. As a result of the evaluation, the Regulatory dose limit was satisfied, and in future studies, internal and external exposure doses were evaluated in consideration of the route of movement, and it is intended to be used as basic data in the field introduction process.

Acknowledgements

This work was supported by the Technology development Program (S3324850) funded by the Ministry of SMEs and Startups (MSS, Korea).

Keywords: Movable radionuclide analysis system, Decommissioning waste, Exposure dose, Workload

A Study on the Method of Screening Radionuclides of Concerns for Kori Unit 1 Decommissioning

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This study presents a methodology to determine the radionuclides of concern that are expected to be found during the final status survey of Kori Unit 1 decommissioning. The methodology involved reflecting the evaluation results of ORIGEN based on reference documents such as NUREG/CR-3474, NUREG/CR-4289, NUREG/CR-0130, WINCO-1191, and representative fuel loading. A list of potential radionuclides of concern was provided by reflecting the list of radionuclides of concern included in the Kori Unit 1 decommissioning plan. To select the radionuclides of concern, we analyzed the approach of US decommissioning plants based on the recommendations of NUREG-1757 Vol.2 Rev.1 and excluded certain radionuclides from the list. The final list of 23 radionuclides of concern was derived by excluding radionuclides that have a short half-life, low specific activity, analytically difficult to measure, inert gases, or naturally occurring radionuclides. This methodology can be applied to other nuclear power plants, such as the Wolsong Nuclear Power Plant, by reflecting the unique characteristics of the reactor.

Acknowledgements

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Keywords: Decommissioning, Radionuclides of concern, Screening, Method

Review of Contents Considered for the Decommissioning Feasibility Designs of Nuclear Facilities

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In order to start decommissioning domestic nuclear facilities, the Final Decommissioning Plan (FDP) must be prepared and approved by the regulatory agency. The contents of domestic FDP consist of 12 chapters, and there is the decommissioning feasibility design that should be described in Chapter 5 as contents to be considered from the construction stage of nuclear facilities.

The design of decommissioning feasibility for nuclear facilities seems to be largely divided into three items. In summary, there are minimization of contaminations to facilities and the environment, easy of dismantling, and minimization of the radioactive waste generation. In addition, the design characteristics to which the ALARA principle is applied in terms of optimizing the exposure dose of workers and residents may also correspond to the decommissioning feasibility design.

The design characteristics for decommissioning feasibility during the period leading up to the design, operation, and decommissioning of nuclear facilities can be listed as the main points as follows. Minimization of facility contamination will include contents related to the leakage of systems and components, minimization of effluents to the environment will involve gaseous and liquid effluents from systems and components to the environment, easy of dismantling will involve history and inspection records during operation, and minimization of radioactive waste generation can be the contents related to the radioactive waste management plans.

The design characteristics of facilities and equipment to meet the ALARA principles can be listed as follows. It means taking into account the benefits and costs of the design improvement plan, and the elimination of unnecessary radiation exposure can be maintained at the exposure dose ALARA, which is in line with the decommissioning feasibility design.

Among the requirements of licensing documents for decommissioning domestic nuclear facilities is the decommissioning feasibility design. This item relates to the design characteristics for decommissioning considered in the construction stage of the facility and should present the effectiveness of measures for them until operation and decommissioning. In this study, the regulatory requirements presented in the construction and operation stage and the contents presented in the U.S. case were reviewed, and it is hoped that it will be used as reference for the preparation of FDP.

Keywords: Decommissioning, Feasibility design, Final decommissioning plan

Arc Efficiency Measurement of Plasma Assisted Milling for SUS 316L

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Metals such as stainless steel and alloy 600 are used as structures and materials in nuclear power plants due to their excellent mechanical properties and heat resistance. And recently thermal and mechanical cutting technologies are being actively researched and developed for dismantling NPP. Among them, the mechanical cutting method has the advantage of less secondary waste generation such as fume and fine dust, but according to the wider the cutting range, the reaction force and the cutting device size are increased. In this paper, plasma assisted milling has been proposed to reduce the reaction force and device size, and the plasma efficiency was measured for SUS 316L. The plasma torch was operated at the level of 3 to 4 kW so that it was heated only without cutting. And the feedrate was set at 150 to 250 mm/min. The test confirmed that the plasma efficiency was 35% about SUS 316L, and it is expected that the numerical analysis using these test results can be used as basic data for plasma assisted milling.

Keywords: Plasma, Transferred type torch, Hybrid cutting, SUS 316L, Plasma assisted milling

Basic Design for Memorialization of Research Reactor TRIGA Mark-II

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In this study, in relation to the demolition of the building as a research reactor, in order to establish a basic design for preparation for relocation and installation of the TRIGA Mark-II, the present conditions such as actual measurements and structural safety were investigated, as well as technologies and cases related to the relocation and installation of cultural properties. Based on this, the basic design for the relocation and installation of cultural assets was established by reviewing the disassembly and transport design of the TRIGA Mark-II and the basic plan for the relocation site. Although the structural safety of the current self-weight of the structure is judged to be reasonable, when lifting the structure, it is necessary to consider a method of lifting the foundation by reinforcing the foundation so that the tensile force can be minimized in the structure. As for the technology to be applied before TRIGA Mark-II, the technology before non-transplacement was confirmed as the most reasonable method in terms of preserving the original form, securing safety, and securing economic feasibility. Among the non-replacement technologies, the methods that can be applied before reactor 1 can be largely classified into three types. The three methods to be reviewed can be largely classified into the traditional rail movement method, the movement method using transport equipment, and the crane movement method. Each required period was calculated from the basic design results, and the modular trailer method was judged to be the most efficient. From the basic design results, the required period for each stage according to the mobile construction method was calculated. Depending on the calculation result, the modular trailer method is judged to be the most efficient. However, the final construction method should be selected according to the detailed design results. Overall, the results obtained through this study suggest that it is possible to create a memorial hall without the previous installation of TRIGA Mark-II if the structure foundation is composed independently of the building foundation after conducting a detailed characteristic investigation on the foundation of the TRIGA Mark-II structure.

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Keywords: TRIGA mark-II, National registered cultural property, Relocation installation, Memorial hall, Decommissioning of nuclear facilities

A Study on Disposal Stability of Vitrified Form

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Radioactive waste generated in various forms needs to be technically stabilized for safe treatment, disposal, and long-term safety. Overseas, research on vitrification with excellent mechanical and physicochemical properties of high-level waste is being actively conducted. Vitrification is a process of converting radioactive waste into a stable and durable glass-like material, which is then safe storage and disposal. The stability of vitrified form is an important concern for environment safety, as any leakage or release of radionuclide could have serious consequences for health and the environment. Therefore, several studies are being conducted on the disposal stability of vitrification of radioactive waste. In order to evaluate the stability of the solidified form, mechanical properties such as density, microhardness, and compressive strength and chemical durability such as leaching properties should be performed. There are several types of leaching test methods to evaluate the chemical durability, which is important in the characterization of solidified forms.

In this study, the leaching test method for chemical durability evaluation and the evaluation results of leaching characteristics according to pH were reviewed.

Keywords: Vitrification, Leaching test, Chemical durability

A Study on the Utilization of Thermal Effluent Discharge From NPPs for Extracting Dissolved Resources in Seawater

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Seawater containing metals such as lithium and manganese is a “treasure trove” of infinite energy resources. Numerous domestic and foreign institutions are developing technologies to economically extract these resources from seawater. One method for extracting metal ions dissolved in seawater is the development of adsorbents with negative functional groups. Generally, adsorbents have adsorption performance that depends on factors such as seawater pH and temperature, but controlling the pH and temperature of seawater is practically impossible. On the other hand, thermal effluent discharged from power plants tends to be slightly higher in temperature than the surrounding environment. Therefore, this study investigates the potential for utilizing power plant effluent to extract dissolved resources in seawater. Results of investigations into several items related to the effluent from the Gori, Wolsong, Hanbit, and Hanul power plants are presented.

Keywords: Energy resource, Seawater, Adsorbent, Thermal effluent discharge

Review on Domestic Technology Development for Extracting Dissolved Resources From Seawater

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To develop technology for extracting energy resources from seawater, we first investigated the research experiences of domestic experts. The survey items included the types of adsorbents that can adsorb dissolved resources in seawater, the subjects of experiments, and the scope of research. We divided the types of adsorbents into organic and inorganic categories and compared their adsorption performance. We also examined how adsorption experiments were conducted using simulated solutions and confirmed whether there were any experiences of conducting experiments in actual seawater. A total of 14 domestic research papers on extracting dissolved resources from seawater were reviewed, excluding topics such as removing dissolved resources from seawater and seawater desalination. This review provides an understanding of domestic research trends and will be helpful in setting directions for future research and development.

Keywords: Energy resource, Seawater, Adsorbent

Efficient Radiochemical Analysis of Carbon-14 in Radioactive Wastes

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Disposal of radioactive waste requires radiological characterization. Carbon-14 (C-14) is a volatile radionuclide with a long half-life, and it is one of the important radionuclides in a radioactive waste management. For the accurate liquid scintillation counter (LSC) analysis of a pure beta-emitting C-14, it should be separated from other beta emitters after extracted from the radioactive wastes since the LSC spectrum signals from C-14 overlaps with those from other beta-emitting nuclides in the extracted solutions. There have been three representative separation methods for the analysis of volatile C-14 such as acid digestion, wet oxidation, and pyrolysis. Each method has its own pros and cons. For example, the acid digestion method is easily accessible, but it involves the use of strong acids and generates large amount of secondary wastes. Moreover, it requires additional time-consuming purification steps and the skillful operators.

In this study, more efficient and environment-friendly C-14 analysis method was suggested by adopting the photochemical reactions via in-situ decomposition using UV light source. As an initial step for the demonstration of the feasibility of the proposed method, instead of using radioactive C-14 standards, non-radioactive inorganic and organic standards were investigated to evaluate the recovery of carbon as a preliminary study. These standards were oxidized with chemical oxidants such as H₂O₂ or K₂S₂O₈ under UV irradiations, and the generated CO₂ was collected in Carbo-Sorb E solution. Recovery yield of carbon was measured based on the gravimetric method. As an advanced oxidation process, our photocatalytic oxidation will be promising as a time-saving method with less secondary wastes for the quantitative C-14 analysis in low-level radioactive wastes.

Keywords: Radiochemical analysis, Carbon-14, Photocatalytic oxidation, In-situ decomposition, UV

Review on Influencing Factors for the Selection of the Waste Routes in a Decommissioning Project

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Kori Nuclear Power Plant Unit 1, which began operating in 1978, is Korea's oldest commercial nuclear reactor. The reactor was permanently shut down in June 2017, and now the decommissioning process has begun. The decommissioning process will generate a significant amount of waste that requires appropriate management to minimize the impact on the environment and human health. And the waste routing, i.e. the activities and logistics for managing the material generated, is a key point in a decommissioning project. It determines the routes from the material inventory to the envisaged material end states.

In this study, we review on several factors for the selection of the waste routes in a decommissioning project. In terms of sustainability, the 'waste hierarchy' should be applied to routing materials from nuclear facilities. According to the waste hierarchy, the preferred end state is reuse or recycling of the waste as material or, more preferably, the avoidance of waste generation. In addition, treatments (such as decontamination and thermal treatment) that can reduce the volumes requiring disposal as radioactive waste should be considered. Another important parameter is the need to secure availability and capacity of waste routes. Short-term bottlenecks or any delay in the removal of the waste from the site often has an impact on other site activities. If possible, at least two alternative waste routes should be identified for the main categories of waste and kept available throughout the decommissioning project. All routes should be direct to the material end state if possible, but it is more important that waste is removed from the site so that other site operations are not impeded.

Keywords: Decommissioning, Waste route, Influencing factor

Review on ASME Standard Guide for Preparing Characterization Plans for Decommissioning Nuclear Facilities

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As an initial part of Kori-1 & Wolsung-1 Unit decommissioning planning, a characterization plan is developed to define the nature, extent and location of contaminants, determine sampling locations and protocols, determine quality assurance objectives for characterization, and define documentation requirements.

The actual characterization of a facility is an iterative process that involves initial sampling according to the characterization plan, field management (such as labeling, packaging, storing, and transport) of the samples, laboratory analysis, conformance to the data quality objectives (DQOs), and then identifying any additional sampling required, refining the DQOs, and modifying the characterization plan accordingly.

The final product of the facility characterization is a document that describes the type, amount, and location of contaminants that will require consideration and removal during the decommissioning operations sufficient to prepare a decommissioning plan.

In this study, implementing a characterization plan, developed in accordance with this standard, will result in obtaining or deriving the above information.

Keywords: Characterization plan, Data quality object, Decommissioning

A Study on Types and Radiological Characterization Methods of Solid Radioactive Waste During Decommissioning in Kori Unit 1

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Kori unit 1, the first PWR (Pressurized Water Reactor) in Korea, was permanent shut down in 2017. In Korea, according to the Nuclear Safety Act, the FDP (Final Decommissioning Plan) must be submitted within 5 years of permanent shutdown. According to NSSC Notice, the types, volumes, and radioactivity of solid radioactive wastes should be included in FDP chapter 9, Radioactive Waste Management. Therefore, in this study, the types depending on generation characteristics and radiological characterization methods and process of solid radioactive waste were analyzed.

Solid radioactive waste depending on the characteristics of the generation was classified into reactor vessel and reactor vessel internal, large components, small metals, spent nuclear fuel storage racks, insulation, wires, concrete debris, scattering concrete, asbestos, mixed waste, soil, spent resins and filters, and dry active waste.

Radiological characterization of solid radioactive waste is performed to determine the characteristics of radioactive contamination, including the type and concentration of radionuclides. It is necessary to ensure the representativeness of the sample for the structures, systems and components to be evaluated and to apply appropriate evaluation methods and procedures according to the structure, material and type of contamination. Therefore, the radiological characterization is divided into concrete and structures, systems and components, and reactor vessel, reactor vessel internal and bioshield concrete.

In this study, the types depending on generation characteristics and radiological characterization methods and process of solid radioactive waste were analyzed. The results of this study can be used as a basis for the preparation of the FDP for the Kori unit 1.

Keywords: Waste classification, Solid radioactive waste, Characterization, Kori unit 1, Decommissioning

A Study on the Development of Methyl Iodine Adsorption Materials Using Waste Concrete and Waste Soil

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Radioactive waste generated during decommissioning of nuclear power plants is classified according to the degree of radioactivity, of which concrete and soil are reclassified, some are discharged, and the rest is recycled. However, the management cost of large amounts of concrete and soil accounts for about 40% of the total waste management cost. In this study, a material that absorbs methyl iodine, a radioactive gas generated from nuclear power plants, was developed by materializing these concrete and soil, and performance evaluation was conducted. A ceramic filter was manufactured by forming and sintering mixed materials using waste concrete, waste soil, and by-products generated in steel mills, and TEDA was attached to the ceramic filter by 5wt% to 20wt% before adsorption performance test. During the deposition process, TEDA was vaporized at 95°C and attached to a ceramic filter, and the amount of TEDA deposition was analyzed using ICP-MS. The adsorption performance test device set experimental conditions based on ASTM-D3808. High purity nitrogen gas, nitrogen gas and methyl iodine mixed gas were used, the supply amount of methyl iodine was 1.75 ppm, the flow rate of gas was 12 m/min, and the supply of water was determined using the vapor pressure value of 30°C and the ideal gas equation to maintain 95%. Gas from the gas collector was sampled to analyze the removal efficiency of methyl iodine, and the amount of methyl iodine detected was measured using a methyl iodine detection tube.

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Keywords: Waste soil, Waste concrete, Methyl iodide, TEDA, Radioactive gases

A Study on the Sampling Method of Activated Structures Around the Reactor During the Decommissioning of Nuclear Power Plants

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To evaluate the characteristics of radioactive waste from permanently shut down nuclear power plants for decommissioning, there is a method of directly analyzing samples and, on the other hand, a computerized evaluation method based on operation history.

Even if the radioactivity of the structures or radioactive wastes in the nuclear power plant is analyzed by the computerized evaluation method, a method of directly analyzing the sample must be accompanied in order to more accurately know the characteristics of the nuclear power plant's radioactive waste material. In order to obtain such samples, we need a way to collect materials from radioactive waste. However, in the case of a permanently shut down nuclear power plant with a long operating history, human access is limited due to radiation of the material.

In this study, we propose a method of remotely collecting samples that guarantees radiation protection and worker safety at the site where radioactive waste is located.

Keywords: Decommissioning, Radioactive waste, Characterization, Sampling

Separation of Hydrogen Isotopes by Membrane Pervaporation of Water

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The separation of hydrogen isotopes is a critical issue in various fields, such as deuterium or tritium production and the treatment of radioactively contaminated water. In this presentation, we describe the pervaporative separation of hydrogen isotopes using proton conductive membranes and underlying separation mechanism. We investigated the H/D separation factors of perfluorosulfonic acid (Nafion) and polybenzimidazole membranes using pervaporation, and found that both membranes exhibited similar separation factors of approximately 1.026. Water permeation flux through the membranes was highly dependent on their thickness and type, and increased with operation temperature. However, the effect of temperature on H/D separation factor was negligible. We also demonstrated the cascade separation of H/D, indicating the potential application of multi-stage operation. We found that surface transport mechanisms such as hydron hopping contributed the most to H/D separation during the pervaporation process of proton conductive membranes.

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Keywords: Tritium, Deuterium, Separation, Membrane, Pervaporation

Development of Inorganic Adsorbent for Tritium Contaminated Water Treatment

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In this study, four technologies were selected to treat river water, lake water, and groundwater that may be contaminated by tritium contaminated water and tritium outflow from nuclear power plants, performance evaluation was performed with a lab-scale device, and then a pilot-scale hybrid removal facility was designed.

In the case of hybrid removal facilities, it consists of a pretreatment unit, a main treatment unit, and a post-treatment unit. After removing some ionic, particulate pollutants and tritium from the pretreatment unit consisting of UF, RO, EDI, and CDI, pure water (2 $\mu\text{S}/\text{cm}$) tritium contaminated water is sent to the main treatment process. In this treatment process, which is operated by combining four single process technologies using an inorganic adsorbent, a zeolite membrane, an electrochemical module and aluminum-supported ion exchange resin, the concentration of tritium can be reduced. At this time, the tritium treatment efficiency of this treatment process can be increased by improving the operation order of four single processes and the performance of inorganic adsorbents, zeolite membrane, electrochemical modules, and aluminum-supported ion exchange resins used in a single process.

Therefore, in this study, as part of a study to increase the processing efficiency of the main treatment facility, the tritium removal efficiency according to the type of inorganic adsorbent was compared, and considerations were considered when operating the complex process.

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Keywords: Tritium, Tritium removal, Adsorption of tritium, Inorganic adsorbent

A Study on Consideration for Selecting Reference Area for Decommissioning of Nuclear Facilities

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After permanent shutdown, contamination existing in nuclear facilities must be removed according to decontamination and dismantling procedures to achieve the target end state. In Korea, Korea Research Reactor (KRR) Units 1, 2 are being decommissioned, and Kori Unit 1 is in the process of reviewing the final decommissioning plan for the start of decommissioning. In order to complete decommissioning of nuclear facilities, it is necessary to satisfy the dose criteria according to the residual radioactivity remaining in the site and buildings.

In the United States, which has a lot of experience in decommissioning, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) is used as a procedure for measuring and analyzing residual radioactivity. In MARSSIM, survey units are classified according to the level of contamination, and the radiation survey procedure and effort can be determined according to the survey unit level. After the radiological analysis and statistical verification of the survey unit, it is decided whether to release the site. At this time, the geographical area to be used as the background level is called the reference area. Therefore, selection of an appropriate reference area is important for accurate residual radioactivity analysis and for the release of the site.

In this study, reference area evaluation cases and domestic decommissioning procedures were analyzed to derive considerations for selecting an appropriate reference area. For example, Zion NPP in the US selected a place outside the boundary of the restricted area unaffected by nuclear power plant operation as a reference area by referring to the meteorological monitoring report. Among Korea's decommissioning procedures, the appropriateness of the reference area can be confirmed through the final status report submitted upon completion of decommissioning. However, since the selection and application of the reference area needs to be reflected during decommissioning, relevant information must be updated through periodic communication between operator and regulatory agency. The results of this study will be used as considerations for selecting a reference area.

Keywords: Decommissioning, Reference area, MARSSIM

Preliminary Approach to Optimize Predisposal Management of Radioactive Waste Through Varying Potential Storage Time

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The amount of waste that contains or is contaminated with radionuclides is increasing gradually due to the use of radioactive material in various fields including the operation and decommissioning of nuclear facilities. Such radioactive waste should be safely managed until its disposal to protect public health and the environment. Predisposal management of radioactive waste covers all the steps in the management of radioactive waste from its generation up to disposal, including processing (pretreatment, treatment, and conditioning), storage, and transport.

There could be a lot of strategies for the predisposal management of radioactive waste. In order to comply with safety requirements including Waste Acceptance Criteria (WAC) at the radioactive waste repository however, the optimal scenario must be derived. The type and form of waste, the radiation dose of workers and the public, the technical options, and the costs would be taken into account to determine the optimal one. The time required for each process affects the radiation dose and respective cost as well as those for the following procedures. In particular, the time of storing radioactive waste would have the highest impact because of the longest period which decreases the concentrations of radionuclides but increases the cost. There have been little studies reported on optimization reflecting variations of radiation dose and cost in predisposal management scenarios for radioactive waste.

In this study, the optimal storage time of radioactive waste was estimated for several scenarios. In terms of the radiation dose, the cumulative collective dose was used as the parameter for each process. The cost was calculated considering the inflation rate and interest rate. Since the radiation dose and the cost should be interconvertible for optimization, the collective dose was converted into monetary value using the value so-called “alpha value” or “monetary value of Person-Sv”.

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Keywords: Predisposal management, Optimization, Storage time, Monetary value of person-sv

Proposal of Economical and Efficient Treatment Facility for Treating Pressurized Heavy Water Reactor Resin Mixture

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Pressurized Heavy Water Reactors (PHWR) have stored ion exchange resins, which are used in deuteration, dehydrogenation systems, liquid waste treatment systems, and heavy water cleaning systems, in spent resin storage tanks. The C-14 radioactivity concentration of PHWR spent resin currently stored at the Wolsong Nuclear Power Plant is 4.6×10^6 Bq/g, which exceeds the limited concentration of low-level radioactive waste. In addition, when all is disposed of, the total radioactivity of C-14, 1.48×10^{15} Bq, exceeds the disposal limit of the first-stage disposal facility, 3.04×10^{14} . Therefore, it is currently impossible to dispose of them in Gyeongju intermediate- and low-level disposal facilities.

As to dispose of spent resins produced in PHWR, C-14 must be removed from spent resins. This C-14 removal technology from the spent resin can increase the utilization of Gyeongju intermediate- and low-level disposal facilities, and since C-14 separated from the spent resin can be used as an expensive resource, it is necessary to maximize its economic value by recycling it.

The development of C-14 removal technology from the spent resin was carried out under the supervision of Korea Hydro & Nuclear Power in 2003, but there was a limit to the C-14 removal and adsorption technology and process. After that, Sunkwang T&S, Korea Atomic Energy Research Institute, and Ulsan Institute of Science and Technology developed spent resin treatment technology with C-14-containing heavy water for the first and second phases from 2015 to 2019 and from 2019 to the present, respectively. The first study had a limitation of a pilot device with a treatment capacity of 10L per day, and the second study was insufficient in implementing the technology to separate spent resin from the mixture, and it was difficult to install on-site due to the enlarged equipment scale.

The technology to be proposed in this paper overcomes the limitations of spent resin mixture separation and equipment size, which are the disadvantages of the existing technology. In addition, since $^{14}\text{CO}_2$ with high concentration is stored in liquid form in the storage tank, only the necessary amount of C-14 radioactive isotope can be extracted from the storage tank and be used in necessary industrial fields such as labeling compound production. Therefore, when the facility proposed in this paper is applied for treating mixtures in spent resin tanks of PHWR, it is expected to secure field applicability and safety, and to reflect the various needs of consumers of labeled compound operators utilizing C-14.

Keywords: PHWR, Spent resin, Microwave, Wolsung, C-14

Introduction of Evaluation and Classification System for Radioactive Soil and Concrete Wastes From NPP Decommissioning

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During decommissioning and site remediation of nuclear power plant, large amount of wastes (including radioactive waste) with various type will be generated within very short time. Among those wastes, soil and concrete wastes is known to account for more than 70% of total waste generated. So, efficient management of these wastes is very essential for effective NPP decommissioning. Recently, BNS (Best System) developed a system for evaluation and classification of soil and concrete wastes from the generation. The system is composed of various modules for container loading, weight measurement, contamination evaluation, waste classification, stacking, storage and control. By adopting modular type, the system is good for dealing with variable situation where system capacity needs to be expanded or contracted depending on the decommissioning schedule, good for minimizing secondary waste generated during maintenance of failed part and also good for disassemble, transfer and assemble. The contamination evaluation module of the system has two sub module. One is for quick measurement with NaI(Tl) detector and the other is for accurate measurement with HPGe detector. For waste transfer, the system adopts LTS (Linear Transfer System) conveyor system showing low vibration and noise during operation. This will be helpful for minimizing scattering of dust from the waste container. And for real time positioning of waste container, wireless tag was adopted. The tag also used for information management of waste history from the generation. Once a container with about 100 kg of soil or concrete is loaded, it is moved to the weight measurement module and then it transfers to quick measurement module. When measured value for radioactivity concentration of Co-60 and Cs-137 is more than 1.0 Bq/g, then the container is classified as waste for disposal and directly transferred to stacking and storage rack. Otherwise, the container is transferred to accurate measurement module. At the accurate module, the container is classified as waste for disposal or waste for regulatory clearance depending on the measurement result of 0.1 Bq/g. As the storage rack has a sections for disposal and regulatory clearance respectively, the classified containers will be positioned at one of the sections depending on the results from the contamination evaluation module. The system can control the movement of lots of container at the same time. So, the system will be helpful for the effective nuclear power plant decommissioning in view of time and budget.

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Keywords: NPP decommissioning, Soil and concrete wastes, Waste classification, LTS conveyor

6분과

방사선환경 및 안전(Oral)

Analysis of Imported Cargo Radiation Monitoring System in Korea

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After the Fukushima nuclear accident in Japan in March 2011, many Koreans were concerned that products exposed to radioactive materials released from the nuclear power plant would be imported into Korea. Systematic radiation monitoring was required for food and daily necessities imported from the nuclear accident area. The need for a legal system to support systematic radiation monitoring was also demanded. The Act on Protective Action Guidelines against Radiation in the Natural Environment was enacted to resolve concerns regarding environmental radiation in Korea in July 2011. According to this law, radiation monitoring equipment has been installed and operated at major airports and ports nationwide.

This paper aims to review the radiation monitoring system of the Korean government comprehensively. The legal system and the legal basis for radiation monitoring of imported cargo conducted by each department were investigated by analyzing the laws and regulations of radiation monitoring for the relevant cargo items. In addition, the current status of radiation monitoring by the government departments was examined to determine how radiation monitoring for imported cargo is performed within the legal system.

The investigation of the current radiation monitoring system for imported cargo in Korea confirmed that radiation monitoring is conducted by classifying cargo items under the jurisdiction of each government department for all imported cargo. However, the reduction in efficiency of radiation monitoring of imported cargoes, unclear legal grounds for radiation monitoring of imported cargo by some departments, the occurrence of overlapping inspections by departments, and the difficult process of issuing the radiation test certificate required for customs clearance by the Korea Customs Service were also identified. As a result of the analysis, it was found that the current radiation monitoring system for imported cargo in Korea ought to be improved, taking into account efficiency, overlapping inspection, legal background, and the difficult process of issuing the certificates.

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Keywords: Environmental radiation, Radiation monitoring system, Imported cargo

High-speed Gamma Imaging System for Nuclear Decommissioning and Waste Management

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Gamma imaging devices that can accurately localize the radioactive contamination could be effectively used during nuclear decommissioning or radioactive waste management. While several hand-held devices have been proposed, their low efficiency due to small sensors have severely limited their application. To overcome this limitation, a high-speed gamma imaging system is under development which comprises two quad-type detectors and a tungsten coded aperture mask. Each quad-type detector consists of four rectangular NaI(Tl) crystals with dimensions of $146 \times 146 \text{ mm}^2$ and 72 square-type photomultiplier tubes (PMTs). The detectors are placed in front and back to serve as scatter and absorber, respectively, for Compton imaging. In addition, a coded aperture mask was fabricated in rank 19 modified uniformly redundant array pattern and placed in front of the scatter for coded aperture imaging. The system offers several advanced features including 1) high efficiency achieved by employing large-area NaI(Tl) crystals and 2) broad energy range of imaging by employing a hybrid imaging combining Compton and coded aperture imaging. The imaging performance of the system was evaluated through experiments in various conditions with different gamma energies and source positions. The imaging system provides clear images of the source locations for gamma energies ranging from as low as 59.5 keV (^{241}Am) to as high as 1,330 keV (^{60}Co). The imaging resolution was within the range of $7.5\text{--}9.4^\circ$, depending on gamma energies, when a hybrid maximum likelihood estimation maximization (MLEM) algorithm was used. The developed system showed high sensitivity, as the ^{137}Cs source at distance, incurring dose rate lower than background level ($0.03 \text{ }\mu\text{Sv/h}$ above background dose rate), could be imaged in approximately 2 seconds. Even under lower dose rate condition (i.e., $0.003 \text{ }\mu\text{Sv/h}$ above background dose rate), the system was able to image the source within 30 seconds. The system developed in the present study broadens the applicable conditions of the gamma ray imaging in terms of gamma ray energy, dose rate, and imaging speed. The performance demonstrated here suggests a new perspective on radiation imaging in the nuclear decontamination and radioactive waste management field.

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Keywords: High-speed gamma imaging system, High sensitivity radiation detector, Hybrid imaging, Nuclear decommissioning, Radioactive waste management

Enhancement of Detection Accuracy in Position-sensitive Plastic Scintillating Fiber-based Sensor Using ANN Model

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In this study, we evaluate artificial neural network (ANN) models that estimate the positions of gamma-ray sources from plastic scintillating fiber (PSF)-based radiation detection systems using different filtering ratios. The PSF-based radiation detection system consists of a single-stranded PSF, two photomultiplier tubes (PMTs) that transform the scintillation signals into electric signals, amplifiers, and a data acquisition system (DAQ). The source used to evaluate the system is Cs-137, with a photo-peak of 662 keV and a dose rate of about 5 μ Sv/h. We construct ANN models with the same structure but different training data.

For the training data, we selected a measurement time of 1 minute to secure a sufficient number of data points. Conversely, we chose a measurement time of 10 seconds for extracting time-difference data from the primary data, followed by filtering. During the filtering process, we identified the peak heights of the gaussian-fitted curves obtained from the histogram of the time-difference data, and extracted the data located above the height which is equal to the peak height multiplied by a predetermined percentage. We used percentage values of 0, 20, 40, and 60 for the filtering.

The results indicate that the filtering has an effect on the position estimation error, which we define as the absolute value of the difference between the estimated source position and the actual source position. The estimation of the ANN model trained with raw data for the training data shows a total average error of 1.391 m, while the ANN model trained with 20%-filtered data for the training data shows a total average error of 0.263 m. Similarly, the 40%-filtered data result shows a total average error of 0.119 m, and the 60%-filtered data result shows a total average error of 0.0452 m. From the perspective of the total average error, it is clear that the more data are filtered, the more accurate the result is.

Further study will be conducted to optimize the filtering ratio for the system and measuring time by evaluating stabilization time for position estimation of the source.

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Keywords: Position estimation, Gamma-ray source, Artificial neural network, Plastic scintillating fiber

Predicting and Preventing Drone Terrorism Using Random Forests Machine Learning Algorithm

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The increasing use of drones in terrorist attacks highlights the need for effective strategies to prevent and respond to drone terrorism. This study uses machine learning approach to identify factors that predict the success of drone terrorism and suggests policy alternatives for preventing such acts. Drone terrorism is becoming increasingly accessible due to advancements in information and communication technology, and events such as North Korea's drone infiltration and the Russia-Ukraine war demonstrate the potential threat of drone attacks on Important National Facilities, including nuclear power plants.

Using the Global Terrorism Database (GTD), this study analyzed drone terrorism incidents that occurred worldwide from 2016 to 2020. The study employed the Random Forest algorithm, which can incorporate multiple factors and their interactions, making it particularly suitable for social science research. The study provides new insights by deriving predictors that were previously overlooked in empirical analyses of drone terrorism.

The findings of this study can aid in the establishment of anti-terrorism policies aimed at addressing the growing threat of drone terrorism. This can include the organization and expansion of the crisis management governance terrorism response council, the creation of a working manual through the partial revision of laws concerning drone terrorism response, and the implementation of anti-drone equipment and systems.

Ultimately, the insights gained from this study can provide development of effective strategies aimed at preventing and responding to drone attacks. The study highlights the importance of proactive measures to mitigate the risks posed by drone technology in the context of terrorism.

Keywords: Drone terrorism, Machine learning, Random forest, New attack means, Complex terrorism, Drone defense system

Comparison Experiment of Environmental Radiation Monitoring With NaI(Tl), LaBr₃, and CZT Detectors Using Unmanned Aerial Vehicle

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Airborne surveys are an essential analysis method for rapid response and contamination identification in the early event of a radiation emergency. On the other hand, airborne surveys are far from the ground, so it is necessary to convert the dose rate at a height of 1 m above the ground. In order to improve the accuracy of the analysis value, a lot of analysis of the measurement data is required. In our previous research, we developed MARK-A1, an instrument for rapid radiation aerial survey in high radiation environment, and MARK-M1, a multipurpose instrument for radiation detection. In this study, a large unmanned aerial vehicle (UAV) was used to conduct airborne surveys using three types of detectors in the Jeju Island environment. The NaI(Tl) detector uses one 3-inch scintillator (38 mm ϕ $\times$ 38 mm H), and the LaBr₃ detector uses two 2-inch scintillators (25 mm ϕ $\times$ 25 mm H). The CZT detector uses a detector with dimensions of (15 mm $\times$ 15 mm $\times$ 7.5 mm). The UAV has a payload of 15 kg (J10, JCH systems Inc. Seoul, Korea). Three different detectors were operated at a constant height of 20 m, 30 m, and 50 m. The flight experiments were performed in the west near Jeju City. Dose rate conversion factors were used to convert the dose rate from the measured spectra, and hovering flights were conducted between 1 and 50 meters to derive altitude correction factors. In this paper, the data measured with each detector in the same area were compared and the differences were derived.

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Keywords: Airborne survey, Radiation measurement, Unmanned aerial vehicle

Study on the Applicability of the Double Sensor Measurement Method to Obtain the Distribution Direction of the Gamma Probe

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Our research team has developed a gamma ray detector which can be distributed over large area through air transport. Multiple detectors (9 devices per 1 set) are distributed to measure environmental radiation, and information such as the activity and location of the radiation source can be inferred using the measured data.

Generally, radiation is usually measured by pointing the detector towards the radioactive sources for efficient measurement. However, the detector developed in this study is placed on the ground by dropping from the drone. Thus, it does not always face toward the radiation source. Also, since it is a remote measurement system, the user cannot know the angle information between the source and detector. Without the angle information, it is impossible to correct the measured value. The most problematic feature is when the backside of the detector (opposite of the scintillator) faces the radiation source. It was confirmed that the measurement value decreased by approximately 50% when the backside of the detector was facing towards the radiation source.

To calibrate the measured value, we need the information that can indicate which part of the detector (front, side, back) faces the source. Therefore, in this study, we installed a small gamma sensor on the backside of the detector to find the direction of the detector. Since this sensor has different measurement specifications from the main sensor in terms of the area, type, efficiency and measurement method, the measured values between the two sensors are different. Therefore, we only extract approximate direction using the variation in the measured value ratio of the two sensors. In this study, to verify the applicability of the detector structure and measurement method, the ratio of measured values that change according to the direction of the source was investigated through MCNP simulation.

The radioactive source was Cs-137, and the simulation was performed while moving in a semicircular shape with 15 degree steps from 0 degree to 180 degrees at a distance of 20 cm from the center point of the main sensor. Since the MCNP result indicates the probability of generating a pulse for one photon, this value was calculated based on 88.6 μCi to obtain an actual count. Through the ratio of the count values of the two sensors, it was determined whether the radioactive source was located in the front, side, or back of the probe.

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Keywords: Denuclearization, Gamma sensor, Plastic scintillator, PIN diode

Strategy for the Mitigation of Radioxenon Emission From Fission-based Radioisotope Production

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Molybdenum-99 (Mo-99) and, its daughter, technetium-99m (Tc-99m) are the most commonly used medical isotope covering more than 85% of the nuclear diagnostics. Currently, majority of Mo-99 supplied in the market is fission-based Mo-99 produced by the fission of U-235 in research reactors. In spite of substitutive production schemes, fission-based Mo-99 is the major source for its significant advantages of high specific activity and large production capacity. The new research reactor (KJRR) is under construction in Gijang, Busan, Korea. The project is aiming 2,000 Ci/week Mo-99 production. For the objective, KAERI has been developed Mo-99 production process using HANARO. Weekly production of 2,000 Ci (100,000 Ci/yr, 6-day calibration) Mo-99 can cover 100% domestic needs, as well as 20% of international demand.

However, overall cost for the fission-based Mo-99 production is continuously increasing. Previously, the most Mo-99 producers used weapon-grade highly enriched uranium (HEU) targets. Recently, the use of HEU in private sector is limited for non-proliferation. As a result, major Mo-99 producers are forced to convert their targets from HEU to low enriched uranium (LEU, 19.75% U-235 enrichment).

The conversion of Mo-99 target caused waste issue. It is not only because of the 50% less yield in production, but also increment of the radioactive waste by 200%. Therefore, designing optimal radioactive waste treatment strategy for fission-based Mo-99 production is becoming more important than ever.

During the process, irradiated LEU targets are dissolved in alkaline solution in hot cells. Fission products other than Mo-99 removed from the solution via series of separation steps. Then Mo-99 is eluted and purified to meet international standard as an active pharmaceutical ingredients (APIs). Radioisotopes of xenon (Xe) and krypton (Kr) generated from the fission of U-235 during the irradiation of the target in the research reactor. Then, the radioactive gas released during the process. The emission of radioactive noble gas from the medical radioisotope production facility can be controlled via delayed release through large charcoal beds.

KAERI developed compact xenon adsorption module with chilled carbon column to meet 5 GBq/day of CTBTO recommendation. Small volume of chilled charcoal can satisfy the guideline, replacing massive gas tank system.

Therefore, development of optimized radioactive gas treatment system for the Mo-99 production is one of the essential piece for the successful construction, licensing and operation of the KJRR project.

Keywords: Radioxenon, Noble gas, Charcoal, CTBTO, Xenon mitigation

Comparison of the Amount of Carbon-14 Radioactive Effluents Discharged From Korean PWRs and PHWRs and Its Resulting Public Dose

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As of 2023, there are a total of 24 nuclear power plants (NPPs) in operation in Korea, with 21 pressurized water reactors (PWRs) and three pressurized heavy water reactors (PHWRs). Korean NPPs discharge radioactive effluents into the environment every year. Radioactive effluents from NPPs contain various radionuclides and materials, including ^3H , ^{14}C , Noble gas, particulates, and iodine

Among the radioactive effluents discharged from Korean NPPs, ^{14}C is a pure beta emitter with a half-life of 5,730 years. The human body can inhale and ingest ^{14}C to cause internal exposure. In addition, the amount of ^{14}C present in the environment is decreasing, but the amount of ^{14}C discharged from NPPs is increasing. ^{14}C discharged to the environment can be inhaled and ingested by the public around NPPs through various pathways. For this reason, it is very important to monitor and manage ^{14}C because it affects the dose of the public around NPPs.

Therefore, this study compared and analyzed the average emissions of ^{14}C discharged from PWRs and PHWRs during the recent 10 years (2012-2021). An average of the public dose due to ^{14}C released from NPPs depending on the reactor types from 2012 to 2021 was also compared. It is inevitable to discharge radioactive effluents while operating NPPs. Korea Hydro & Nuclear Power (KHNP) manages and monitors radioactive effluents released into the environment. According to a survey and analysis of ^{14}C discharged from PWRs and PHWRs and the average dose of the public over the recent 10-year (2012-2021) around Korean NPPs, ^{14}C released from PWR accounted for 3.1% of the total discharge but accounted for more than 93.67% of the total public dose. In addition, ^{14}C discharged from PHWRs accounted for 1.12% of the total discharge, but its resulting dose to the public accounted for more than 83.87% of the total public dose. As a result of analyzing the public dose due to ^{14}C from 2012 to 2021, it was gradually increasing every year. Based on these results, monitoring and managing ^{14}C discharge and its resulting doses to the public are important in the future.

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Keywords: NPP, Radioactive effluents, Carbon-14, Discharge, Public dose

6분과

방사선환경 및 안전(Poster)

Development of a Radiation Prediction Model Using the Kriging Method

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In the event of a radioactive release, it is essential to quickly detect and locate the source of the release, as well as track the movement of the plume to assess the potential impact on public health and safety. Fixed monitoring posts are limited in their ability to provide a complete picture of the radiation distribution, and the information they provide may not be available in real-time. This is why other types of monitoring systems, such as mobile monitoring, aerial monitoring, and personal dosimeters, are also used in emergency situations to complement the information provided by fixed monitoring posts.

Also, the monitoring system can be improved by using the Kriging technique, which is one of the interpolation methods, to predict the radiation dose in the relevant districts. This can be achieved by utilizing both the GPS information and the radiation dose measured at a particular point. The Kriging method involves estimating the value between different measurement points by considering the distance between them.

The model used GPS and radiation data that were measured around the Hanbit NPP. The data were collected using a radiation measuring detector on a bus that traveled around the NPP area at 2-second intervals for one day. From the collected data, 200 data points were randomly selected for analysis, excluding the data measured at the bus garage out of a total of 16,550 data points. The average dose of the daily measurement data was 117.94 nSv/h, and the average dose of the 200 randomly extracted data was 119.17 nSv/h. The GPS and radiation dose data were utilized to predict the radiation dose around the Yeonggwang area where the Hanbit NPP is located.

In the event of an abnormal release of radioactive material, it can be difficult to accurately determine the dose unless a monitoring measurement point is present. This can delay the rapid evacuation of residents during an emergency situation. By utilizing the Kriging model to make predictions, it is anticipated that more accurate dose predictions can be generated, particularly during accident scenarios. This can aid in the development of appropriate resident protection measures.

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Keywords: Kriging, Radiation dose, Prediction, Accident

Improvement of Gamma-ray Analysis Procedures for Reducing Human Error and Workload Using Automation

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Radioactive waste can be classified according to the concentration level for radionuclides, and the disposal method is different through the level. Gamma analysis is inevitably performed to determine the concentration of radioactive waste, and when a large amount of radioactive waste is generated, such as decommissioning nuclear facilities, it takes a lot of time to analyze samples. The performance of a lot of analysis can cause human errors and workload. In general, gamma analysis is performed using by HPGe detector. Recently, for convenience of analysis, commercial automatic sample changers applicable to the HPGe detectors were developed. The automatic sample changers generate individual analysis reports for each sample. In this study, gamma analysis procedure was improved using the application of the automatic sample changer and the automated data parsing using by Python. The application of automatic sample changers and data parsing technique can solve the problems. The human errors were reduced to 0% compared to the previous method by improving the gamma analysis procedure, and working time were also dramatically reduced. This automation of analysis procedure will contribute to reducing the burden of analysis work and reducing human errors through various improvements.

Keywords: Gamma analysis, HPGe, Automatic sample changer

Study of Pre-Classification Method for Clearance Level Through Net Peak Area Ratio in Natural Uranium-Contaminated Soil

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Natural uranium-contaminated soil in Korea Atomic Energy Research Institute (KAERI) was generated by decommissioning of the natural uranium conversion facility in 2010. Some of the contaminated soil was expected to be clearance level, however the disposal cost burden is increasing because it is not classified in advance. In this study, pre-classification method is presented according to the ratio of naturally occurring radioactive material (NORM) and contaminated uranium in the soil. To verify the validity of the method, the verification of the uranium radioactivity concentration estimation method through γ -ray analysis results corrected by self-absorption using MCNP6.2, and the validity of the pre-classification method according to the net peak area ratio were evaluated.

Estimating concentration for ^{238}U and ^{235}U with γ -ray analysis using HPGe (GC3018) and MCNP6.2 was verified by α -spectrometry. The analysis results of different methods were within the deviation range. Clearance screening factors (CSFs) were derived through MCNP6.2, and net peak area ratio were calculated at 295.21 keV, 351.92 keV(^{214}Pb), 609.31 keV, 1120.28 keV, 1764.49 keV(^{214}Bi) of to the 92.59 keV. CSFs for contaminated soil and natural soil were compared with U/Pb ratio. CSFs and radioactivity concentrations were measured, and the deviation from the 60 minute measurement results was compared in natural soil. Pre-classification is possible using by CSFs measured for more than 5 minutes to the average concentration of ^{214}Pb or ^{214}Bi in contaminated soil.

In this study, the pre-classification method of clearance determination in contaminated soil was evaluated, and it was relatively accurate in a shorter measurement time than the method using the concentrations. This method is expected to be used as a simple pre-classification method through additional research.

Keywords: Radioactive waste, Natural uranium-contaminated soil, Radioactivity estimation, Clearance, Net peak area ratio

Validation of the Reliability of Body Residual Radioactivity Measuring System

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Radiation workers who handle radioisotopes, radioactive waste, nuclear material etc. may be contaminated with radioactive material due to inhalation, resulting in internal radiation exposure. For preventing radiation damage and monitoring the exposure of workers, KAERI operates a Body Radiation Measurement Laboratory.

According to Article 5 of the Nuclear Safety and Security Commission (NSSC) Notice No. 2017-77, “Regulation on Measurement and Calculation of Internal Radiation Dose,” The nuclear energy-related business operator with workers etc. shall establish and operate procedures and methods including the following Subparagraphs to secure the reliability of measurement of the internal radiation dose : operation and calibration of measuring instrument, inspection procedures, uncertainty of measurement, lower limit of detection and geometric configuration used for measurement. In accordance with the provision, Whole Body Counter utilized in the Body radiation Measurement Laboratory has periodic calibration / QA procedures to ensure reliability.

This paper performed reliability validation of the measurement system of the Body Radiation Measurement Laboratory in the KAERI based on the performance criteria for radio-bioassay criteria presented in ISO 28218 and ANSI HPS N13.30-2011(R2017). The first criteria is MTL (Minimum Testing Level). ISO 28218 provides MTLs for each measurement category, type and nuclide. For reliable results, it is recommended to use calibration sources with higher radioactivity than the values given. The MTL for fission products in total body counting is 3 kBq and for the last 3 years the laboratory has been using sources of 6-7 kBq (Co-60, Cs-137 etc.). The second criteria is RMSE (Root Mean Square Error). It is a measure of total error defined as the square root of the sum of the square of the relative precision (SB) and the square of the relative bias (Br). The RMSE shall be lower than or equal to 0.25. The largest RMSE in the last 3 years is 0.12, and average value is 0.065, which meets the criteria.

In this study, we verified the reliability of the radioactivity measurement system (WBC) based on the radio-bioassay standards presented in ISO 28218 and ANSI HPS N13.30-2011(R2017). The values were obtained using 3 years of calibration count data, and it was found that both MTL, RMSE for each nuclide met the standards with a large margin of error and were in good operating condition. This study can be applied to the maintenance, performance check, and reliability verification of similar in vivo radio-bioassay methods.

Keywords: In vivo radio-bioassay, Internal Exposure, MTL, RMSE, Relative Bias, Relative Precision

A Study on Overseas Cases of Radwaste System Evaluation for Continued Operation

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Among domestic Nuclear Power Plants (NPPs), there are a total of 10 nuclear power plants whose operating license expires by 2030, excluding Kori unit 1 and Wolsong unit 1, which are permanently shut down. Continued operation of these nuclear power plants is being reviewed as a government task. For continued operation, nuclear power plant owners must prepare periodic safety review and other evaluation reports to receive reviews to maintain safety even during continued operation. In the safety evaluation of NPP, it is important to refer to overseas cases and operation experiences. In this study, the matters of radioactive waste management for continued operation of NPP was considered by analyzing the safety evaluation reports and safety enhancements of license renewal of NPP in USA. Radioactive waste generated from NPPs can be classified into solid, liquid, and gaseous states. Radioactive waste generated during the operation and maintenance of power plants is classified, stored and treated in the radioactive waste management system according to the source. Equipment and monitors related to radioactive waste management are continuously operated, managed, inspected according to standards and maintain their original functions. Various activities to reduce the generation and emission of radioactive waste from NPPs are performed.

After reviewing the NRC's safety evaluation report on the application documents for license renewal of US NPPs (Sequoyah, Byron and braidwood) the evaluation details and matters requiring enhancement for the radioactive waste management system were confirmed. As a major check, selective leaching occurred in the body of the gray cast iron valve and the heat exchanger shell containing the copper alloy exposed to the radioactive waste liquid. Selective leaching causes loss of material and may interfere with the original function of the facility, so management is required.

For the safe operation and management of NPPs, it is important to refer to overseas cases and experiences. Among the safety evaluations for the continued operation of domestic NPPs, in the field of the radioactive waste management system, if the case of the US NPP is referred to, the review by the regulatory body and the action taken by the licensee will be more efficient.

Keywords: Continued operation, Periodic safety review, Radwaste management system, Selective leaching

A Study on the Periodic Safety Review Process of Domestic and Overseas NPP

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Around the world, Nuclear Power Plants (NPPs) have been operated since the 1950s and are used as a major power source. In Korea, Kori unit 1 started commercial operation for the first time in 1978, and as of 2023, 25 units of NPPs are in operation. NPPs produce electricity for about 40 to 60 years after receiving an operating license, and after securing safety through a safety evaluation, the operating period is extended. NPPs that operate for a long time are systematically evaluated for safety at regular intervals through Periodic Safety Review (PSR) recommended by the IAEA. In Korea, PSR has been introduced and performed since 2000. This study reviewed the process of the PSR by comparing with the international PSR procedure.

The PSR process is established through the IAEA SSG-25 document and proceeds in the order of establishment of basis document - individual factor evaluation - global assessment - integrated improvement plan. In Korea, PSR is carried out in a similar process, but there are some differences from the IAEA's procedure. The safety factor review is conducted under the agreement of basis document between the licensee and the regulatory body, but the prior agreement procedure with the regulatory body is not reflected in Korea. As a result, if the licensee and the regulatory body have different opinions on the current licensing basis and the modern safety standards after the evaluation is performed, a difference may occur in the review results and safety enhancement items, which may lead to inefficient PSR progress.

PSR is conducted for the continuous safe operation and management of NPPs, and it is important to refer to overseas standards and cases. Although procedures, guidelines, and regulatory requirements are in place in Korea, continuous review and improvement are required. It is necessary to improve procedures such as basis document and global assessment in order to more efficiently carry out PSR evaluation by regulatory agency and licensee's safety enhancement actions of domestic NPPs

Keywords: Periodic safety review, Basis document, Global assessment

Suggestion on Screening-out Method of Radionuclides to be Considered for the Offsite Consequence Analysis of Nuclear Power Plants

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As a result of various generation, transmutation, and decay schemes, a wide variety of radionuclides exist in the reactor prior to accident occurrence. Considering all of the radionuclides as the accident source term in an offsite consequence analysis will inevitably take up excessive computer resources and time. Calculation time can be reduced with minimal impact on the accuracy of the results by considering only the nuclides that have a significant effect on the calculation among the potential radioactive sources that may be released into the environment.

In earlier studies related to offsite consequence analysis, it is shown that the principal criteria for the radionuclide screening applied are as follows; radionuclide inventory in the reactor, radioactive half-life, radionuclide release fraction to the environment, relative dose contribution of nuclides within a specific group, and radiobiological importance. As a result, it is confirmed that 54, 60, and 69 nuclides are applied to the risk assessment performed in WASH-1400, NUREG-1150, and SOARCA (State-of-the-Art Reactor Consequence Analyses) project in the United States, respectively. In addition, in this study, the technical consultations with domestic and foreign experts were carried out to confirm details on criteria and process for screening out radionuclides in offsite consequence analysis.

In this paper, based on the literature survey and technical consulting, we derived the screening process of selecting a list of radionuclides to be considered in the offsite consequence analysis. The first step is to eliminate radionuclides with little core inventory (less than specific threshold) or very short half-lives. However, important decay products of radionuclides that have short half-lives should not be excluded by this process. The next step is to further eliminate radionuclides by considering contribution to offsite impact, which is defined as a product of radioactivity released to the environment (i.e. 'inventory in the reactor' times 'release fraction to offsite') and comprehensive dose (or risk) coefficient taking into account all exposure pathways to be included. The final step is to delete isotopes that contribute less than certain threshold to any important dose metric through additional computer runs for each important source term.

Even though it is presumed that this process is applicable to existing light water reactors and the set of accidents that would be considered in PSA, some of the assumptions or specific recommendations may need to be reconsidered for other reactor types or set of accident categories.

Acknowledgements

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Keywords: Radionuclide screening, Screening criteria, Offsite consequence analysis, Level 3 probabilistic safety assessment (PSA)

Review of Microbiologically-influenced Corrosion Monitoring Method for Aging Management of Fuel Oil System

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At Nuclear Power Plant (NPP), aging management is performed as part of the Periodic Safety Review (PSR) in accordance with the Nuclear Safety Act. The purpose of the aging management program (AMP) is to manage the integrity of structures, systems and components (SSCs) in NPPs over time and use. Through this, aging deterioration is mitigated to increase equipment life and secure long-term operation safety.

Fuel Oil Chemistry is one of the AMPs. Through this program, aging management is performed for storage tanks, piping and other metal components that contact with diesel fuel oil. The program is focused on managing loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion (MIC) and fouling that leads to corrosion of the diesel fuel tank internal surfaces. The fuel oil aging management method currently applied to NPPs in Korea measures the concentration of water and particulate contamination in the oil, analyzed the trend, and periodically cleans and inspect the inside of tanks. Among them, in monitoring MIC, a direct analysis and monitoring of the amount of microorganisms may be more effective. In this study, a method for improving the MIC monitoring system for diesel fuel oil systems was reviewed by reviewing reference documents including NUREG 1801 and examining the methods actually applied in US NPPs.

Keywords: Periodic safety review (PSR), Aging management program (AMP), Fuel oil chemistry, Microbiologically-influenced corrosion (MIC)

Preliminary Experiment of Radioactive Caesium Analysis in Seawater Using Ammonium Phosphomolybdate Coprecipitation

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Cs-137, a radioactive isotope of caesium, is a commonly occurring fission product that is generated during the nuclear fission of U-235 and other fissionable isotopes in both nuclear reactors and weapons. Due to its long half-life of about 30 years and propensity to accumulate in sediments and marine organisms, Cs-137 is considered a major radionuclide for environmental radioactivity monitoring. In April 2021, as the Japanese government decided to discharge Fukushima contaminated water into the sea, the monitoring of marine radioactivity in South Korea has become increasingly significant.

In this study, as an initial step towards establishing a standardized procedure for analyzing radioactive caesium in seawater, the radioactivity of Cs-137 was analyzed on a 2 L of seawater spiked with 10 Bq of Cs-137 standard solution supplied by KRISS. The seawater was collected from Im-nang Beach, situated at a distance of approximately 2 kilometers from DIRAMS. The radioactivity of Cs-137 in seawater was determined according to the improved AMP procedure presented by M. Aoyama in 2000. The seawater was pretreated using Ammonium Phosphomolybdate (AMP) coprecipitation, which has a high selectivity for caesium ($K_d = \sim 5500$), and the activity of Cs-137 was determined by gamma-spectroscopy and subsequently corrected via the weight yield.

The weight yield of the dried AMP/Cs compound was more than 93%. For the gamma-spectroscopy analysis, the AMP/Cs compound was dissolved in a cylindrical U8 beaker with NaOH to ensure that its shape and volume were consistent with the CRM (KRISS, 221U890-1) used to calibrate the detector. The dissolved compound was then positioned directly onto the detector housing and subjected to a measurement duration of 80,000 seconds utilizing a p-type HPGe (Ortec, GEM60) with a relative efficiency of 54%. The activity of Cs-137 was determined to be 10.81 Bq, confirming the reproducibility of the AMP coprecipitation and weight yield methods.

The present experiment was carried out using a 2 L sample, but a large volume of seawater would be required to achieve a sufficient minimum detectable activity (MDA) for Cs-137 in natural seawater. Thus, a standardized procedure for analysis of radioactive caesium in natural seawater will be established through the analysis of a large volume of seawater in future studies.

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Keywords: Seawater, Radioactive caesium, Ammonium phosphomolybdate, Coprecipitation, Weight yield

Analysis of Internal Exposure Evaluation Methodology by Aquatic Foods Ingestion in the Maritime Radioactive Waste Transport Sinking Accident

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As nuclear power plants are operated in Korea, low and intermediate-level radioactive wastes and spent nuclear fuels are continuously generated. Due to the increase in the amount of radioactive waste generated, the demand for transportation of radioactive wastes in Korea is increasing. This can have radiological effect for public and worker, risk assessment for radioactive waste transportation should be preceded. Especially, if the radionuclides release in the ocean because of ship sinking accident, it can cause internal exposure by ingestion of aquatic foods. Thus, it is necessary to analyze process of internal exposure due to ingestion. The object of this study is to analyze internal exposure by ingestion of aquatic foods. In this study, we analyzed the process and the evaluation methodology of internal exposure caused by aquatic foods ingestion in MARINRAD, a risk assessment code for marine transport sinking accidents developed by the Sandia National Laboratory (SNL). To calculate the ingestion internal exposure dose, the ingestion concentrations of radionuclides caused by the food chain are calculated first. For this purpose, MARINRAD divide the food chain into three stages; prey, primary predator, and secondary predator. Marine species in each food chain are not specific but general to accommodate a wide variety of global consumer groups. The ingestion concentrations of radionuclides are expressed as an ingestion concentration factors. In the case of prey, the ingestion concentration factors apply the value derived from biological experiments. The predator's ingestion concentration factors are calculated by considering factors such as fraction of nuclide absorbed in gut, ingestion rate, etc. When calculating the ingestion internal exposure dose, the previously calculated ingestion concentration factor, consumption of aquatic food, and dose conversion factor for ingestion are considered. MARINRAD assume that humans consume all marine species presented in the food chain. Marine species consumption is assumed approximate and conservative values for generality. In the internal exposure evaluation by aquatic foods ingestion in this study, the ingestion concentration factor considering the food chain, the fraction of nuclide absorbed in predator's gut, ingestion rate of predator, etc. were considered as influencing factors. In order to evaluate the risk of maritime transportation reflecting domestic characteristics, factors such as domestic food chains and ingestion rate should be considered. The result of this study can be used as basis for risk assessment for maritime transportation in Korea.

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Keywords: Maritime transportation, MARINRAD, Internal exposure

Analysis of Safety Management Standards and Plans for NORM Waste

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Natural radionuclides-containing substances (NORM) contain natural radionuclides and cause radiation exposure. In Korea, safety management measures were needed to deal with and dispose of radon mattresses containing monazite in relation to such NORM. However, there is no clear safety management system related to NORM waste in Korea. In order to manage this reasonably and systematically, it is necessary to investigate and analyze standards and management measures related to the treatment and disposal of NORM waste. Therefore, this study investigated and analyzed the exemption and clearance level of NORM waste regulations in international organizations and foreign countries. IAEA GSR Part 3, 2013/59/Euratom, ANSI/HPS N13.53, CRCPD SSR CR Part N, and ARPANSA Publications 15 safety management regulations were analyzed to investigate safety management standards for NORM waste. The exemption and clearance level in international organizations and foreign countries were compared and analyzed based on radioactive concentration and dose. In addition, the management measures proposed for each literature were also investigated. As a result of the analysis, IAEA GSR Part 3 applied 1 mSv as a regulatory exemption level, 1 Bq/g for uranium and thorium series as a clearance level, and 10 Bq/g for K-40 nuclides. The IAEA recommends a differential approach to the potential and scale of exposure. The EU applied 1 Bq/g to uranium and thorium families and 10 Bq/g to K-40 nuclides for both regulatory exemption and clearance levels. The EU recommended that it be managed in proportion to the scale and likelihood of exposure as a result of the action. It is analyzed that this is similar to the IAEA's management plan. In the United States, there was no single federal government radioactive concentration and dose for NORM management. The management plan differed in management status and level from state to state, and K-40 was excluded from regulation unless it was intentionally enriched. In the case of Australia, the radioactive concentration of uranium and thorium was 1 Bq/g as a standard for regulatory exemption and 1 mSv as a dose. As a management plan, it was suggested to dispose of waste by means of accumulation, dilution/dispersion, and reclamation. It was also suggested that the scale of exposure, like international organizations, take into account the possibility. The results of this study are believed to be used as basic data for presenting domestic NORM waste treatment and disposal methods in the future.

Acknowledgements

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Keywords: NORM, Safety management, NORM waste, Management plan

Uncertainty Evaluation of Deep Dose Conversion Coefficients for an X-ray Irradiation System

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Since 2018, Central Research Institute of Korea Hydro & Nuclear Power (KHNP-CRI) has been operating an X-ray irradiation system with a maximum voltage of 160 kV and 320 kV X-ray tube to test personal dosimeters in accordance with ANSI N13.11-2009 “Personnel Dosimetry Performance-Criteria for Testing”. This standard requires that dosimeters for the photon category testing be irradiated with the X-ray beams appropriate to the ISO beam quality requirements. KHNP-CRI has implemented the fourteen X-ray reference radiation beams in compliance with ISO-4037-1, 2, and 3.

When installing the X-ray irradiation system, KHNP-CRI evaluated the uncertainties of dose conversion coefficients for deep and shallow doses, based on “Catalogue of X-ray spectra and their characteristic data – ISO and DIN radiation qualities, therapy and diagnostic radiation qualities, unfiltered X-ray spectra” published by Physikalisch Technische Bundesanstalt (PTB).

A CdTe detector (X-123, AMPTEK) with disk type collimators made of tungsten was used to acquire X-ray spectra. The detector was located at 1 m from the center of the target material in the X-ray tubes.

Six uncertainty factors for the dose conversion coefficients for the fourteen X-ray beams were chosen as follows; the minimum and maximum cut-off energies $E_{\min}$ and $E_{\max}$, the air density (ρ), the accuracy of the high-voltage of the X-ray tube, statistics of the pulse height spectra and the unfolding method. For example, uncertainty of each quantity for a HK30 beam was calculated to be 0.3%, 2.32%, 0.19%, 1.25%, and 0.13%, and 0.18%, respectively.

The combined standard uncertainty for the deep dose conversion coefficient of the HK30 beam was calculated to be 2.67%. The coverage factor corresponding to a 95 percent confidence interval was obtained as $k = 1.8$ using a Monte Carlo method, which is slightly lower the coverage factor of $k = 1.95$ for a Gaussian distribution. This seems to result from that two dominant uncertainties, the unfolding uncertainty and minimum cut-off energy uncertainty, follow a rectangular distribution.

Keywords: X-ray irradiation system, Dose conversion coefficient, Uncertainty, Coverage factor

Preliminary Study on the Effects of Abnormal Events to Radioactive Effluent Discharges From NPPs Using K-IRED Database

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After the Fukushima nuclear accident in Japan, concerns have increased about radioactive releases from nuclear power plants (NPPs) into the environment. Analysis of annual radioactive effluent release reports (ARERRs) shows that from 2000 to 2020, abnormal releases of radioactive effluent occurred in 703 out of 1,323 Reactor·years in the United States, accounting for 53% of the total number of reactors in 63 PWRs. Furthermore, when examining incidents and malfunctions recorded in Korea's Operational Performance Information System of Nuclear Power Plant (OPIS) during the same period, it can be estimated that abnormal releases occurred in 9 out of the 324 Reactor·years in PWRs and PHWRs.

Meanwhile, database on radioactive releases from NPPs worldwide was collected, and events of abnormal/unplanned releases were investigated. Based on the data collected from 195 NPPs in 8 countries (South Korea, the United States, Japan, France, the United Kingdom, Germany, Spain, and Canada) over a period of 21 years, totaling 4,607 Reactor·years, a program called K-IRED (KHU-Integrated Radioactive Effluent Database) was developed using MS Access. Using K-IRED, three methodologies have been developed to predict abnormal events based on the annual radioactive releases for each NPPs and radionuclide (or radionuclide group).

Three newly developed methodologies were applied to the 63 NPPs (1,323 Reactor·years) in the United States, categorized by radionuclides (or radionuclide groups). Assuming an increase in radioactive effluent due to abnormal events, the annual increase rate of radioactive effluent was calculated for each methodology and the results were analyzed. The optimal methodology among the three was derived, and the applicability of predicting abnormal events in other NPPs beforehand was examined. Therefore, by predicting abnormal or unplanned releases from NPPs to the environment in advance, it is possible to prevent accidents and reduce public concerns, as suggested by results of this study.

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Keywords: Nuclear power plants, Radioactive effluents, Abnormal event, KHU-Integrated Radioactive Effluent Database (K-IRED)

Study on the Spectral Features of the Scintillation Detector Seeded With Calibration Source

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Detectors used for nuclear material safeguards activities are using scintillator detectors to quickly calculate the uranium enrichment at various nuclear material handling facilities. In order to measure the uranium enrichment, a region of interest is set around 185.7 keV which is the main gamma emission energy of uranium-235 in which the proportional relationship between the amount of uranium-235 and the net count is used. It is necessary to perform channel/energy calibration that a specific channel of the multi-channel analyzer is set to 185.7 keV. Most detector manufacturers have a built-in calibration source so that it is automatically performed when the detector starts to operate. In addition, the scintillator detector requires attention because the channel/energy gain may change depending on the ambient temperature so that a calibration source is used to compensate for this.

In this paper, the spectral features are examined from among the scintillator detectors seeded with calibration sources used for safeguards activities. For this purpose, FLIR's Identifinder-2 R400 T2 model and Canberra's NAID model were used. HM-5 contains about 15nCi of Cs-137 and a photoelectric peak occurs at 662.1 keV. NAID contains about Am-241 of 55 nCi which alpha decays and subsequently emits gamma rays of 59.5 keV and 26.3 keV. The major difference among the detectors occurs in the background spectrum due to the difference in the source. From that kind of spectral features, it can be confirmed that the equipment is operating properly only when the spectrum by the corresponding calibration source is accurately known.

The results of this study will enable a better understanding of the characteristics of scintillator detectors used for uranium enrichment analysis. Therefore, it is expected to be used as basic research for related software utilization as well as development in the future.

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Keywords: Uranium enrichment, Spectral features, Built-in calibration sources, HM-5, NAID

A Review of Possible Risks for the Decommissioning Projects of Nuclear Power Plants

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The decommissioning of the Nuclear Power Plant (NPP) is a long-term project of more than 15 years and will be carried out as a project, which will require project management skills accordingly. The risk of decommissioning project is a combination of many factors such as the decommissioning plan, the matters licensed by the regulatory agency, the design and implementation of dismantling, the dismantling plan and organization, and stakeholders. There will be some difficulties in risk management because key assumptions about many factors and the contents of major risks should be well considered. Risk management typically performs a series of processes ranging from identification and analysis to evaluation. In order to analyze and evaluate risks here, identification of potential risks is the first step, and in order to reasonably select potential risks, various factors mentioned should be considered. Therefore, the purpose of this study is to identify possible risks that should be considered for the decommissioning project in various aspects.

The risk of the decommissioning project can be defined using the hazard keyword, and the risk family presented in the IAEA safety series can also be referred. It would be better to approach the radiological or non-radiological risks that may occur in the dismantling work with the hazard keyword, and if the characteristics of the decommissioning project are reflected, it would be a good idea to approach it on a risk family basis. There are 10 top risks in the risk family, 25 risks at the level 2 and 61 risks at the level 3 are presented. It may be complex to consider these hazards and risks recommended as risk families at the same time, so using the results of safety evaluation as input data for risk identification can be a reasonable approach.

Therefore, this study intended to derive the possible risks of the decommissioning project based on the risk family structure. At this point, the reflection of the safety assessment results was intended to be materialized by considering the hazards checklist. As a result, this study defined and example of 38 possible risks for the decommissioning project, considering the 10 top risk family and lower level risk categories. This result is not finalized, and it will be necessary to further strengthened through expert workshops or HAZOP in the future.

Keywords: Decommissioning, Risks, Hazards, Risk management, Project risks

Comparison of Exposure Pathways to Residents During Operation and Decommissioning of Nuclear Power Plants

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One aspect of securing safety from the operation of Nuclear Power Plants (NPPs) is to evaluate the impact on residents at the facility's exclusive area boundary to confirm that the radiological risk is below the allowable level. Normally, the risks from gaseous and liquid effluents are evaluated during the operation of facilities. Meanwhile, in order to be approved for the decommissioning plan, the environmental risks caused by activities during dismantling is also evaluated. Therefore, this study aims to investigate the exposure pathways considered in evaluating the risks to nearby residents from the operation and decommissioning of nuclear facilities and to examine the differences.

The emission rate by radionuclide is calculated by evaluating the amount of leak from nuclear fuel during the operation of the facility through design data of the NPP. Each of the liquid and gaseous effluents is calculated, and the exposure dose received by nearby residents is calculated by considering the exposure pathways with these emission rates.

In order to initiate the decommissioning of nuclear facilities, approval of the Final Decommissioning Plan (FDP) must be obtained. The FDP chapter shall describe the results of the environmental impact assessment of the decommissioning. It will not differ significantly in the exposure pathways during operation. However, the decommissioning of nuclear facilities is ultimately to remove Systems, Structures, and Components (SSCs) and to remove the regulation of the Nuclear Safety Act by ensuring that sites and remaining buildings meet the criteria for the license termination. In terms of release and reuse of nuclear facilities, the exposure dose to be considered in evaluating the dose can be considered for two main types: the site and the remaining building.

The factors affecting the exposure pathways considered in assessing the environmental impacts considered in the operation and decommissioning of nuclear facilities are due to gaseous and liquid effluents. However, the difference should reflect the impact of NPP operations and decommissioning activities when evaluating the amount of radionuclides released by these effluents. Decommissioning should consider the impact after decommissioning, which is the effect of the receptor by radionuclides remaining on the site and in the remaining buildings. At this time, the effects of the source from the soil and the source from the surface of the building should be considered for the external and internal exposure pathways.

Keywords: Decommissioning, Exposure pathways, DCGL, Environmental impact assessment

Considerations to be Taken Into Account in the Industrial Risk Assessment Process for Dismantling Activities

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For licensees who face the decommissioning project for the first time, even if they can utilize their experience in operation, they should be well prepared and assessed for the risks of dismantling activities reflecting the characteristics of decommissioning. This can be included in the risk management of the decommissioning project, but what we want to discuss in this study is the evaluation of the industrial risk of the actual work before the dismantling work is carried out. We would like to focus more on the review of dismantling activities subject to industrial risk assessment and a series of processes for risk assessment.

The dismantling work plan will need to obtain approval from the supervisory department before work on the Systems, Structures, and Components (SSCs) can be carried out. At this time, risk assessment may be included among many safety-related required documents, which are divided into radiological and non-radiological risks. The target activities at Level 1 level can include preparation for dismantling and maintenance of facilities, dismantling big components, removing the contamination of concrete structures, managing radioactive waste, etc. In addition, it can be composed of preparation work, removal of connections, lifting/installation, cutting, radiation/radioactivity measurement, and withdrawal as detailed work stages of each item's activities. For domestic nuclear decommissioning projects, two major performance organizations, licensees and contractors, must be considered. Regarding risk assessment, the licensee will have a supervisory department controlling decommissioning activities and an HSE department at the site, and a process will need to be established in consideration of the contractor's work organization. Therefore, activities in the risk assessment process may be established.

In this study, risk assessment was reviewed as safety-related matters to be considered when carrying out the dismantling work. Safety-related risk assessment is a necessary procedure for performing practical dismantling activities, and this should be considered well in advance. Therefore, work activities and criteria were established for risk assessment, and the performance process was assumed to apply them. In terms of the performance organization and the responsibilities and roles of the processes to be performed by each organization were constructed, and this can be referred to in the process of preparing for the decommissioning project.

Keywords: Decommissioning, Industrial risk, Project management, Hazard

Factors to Consider for Estimating Worker Exposure Doses of Nuclear Decommissioning Activities

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Safety-related items in the decommissioning Nuclear Power Plants (NPPs) can largely consider safety for workers and residents. At this time, the effects of radioactive contamination on the Systems, Structures, and Components (SSCs) are caused by the performance of work related to Decontamination and Dismantlement (D&D) activities. Classification according to dismantling activities will be important, and the decay factor of radionuclides and the impact of contaminations due to plant characteristic (thermal and electrical capacity) in estimation of exposure dose from such activities will be considered compared to other overseas NPPs. Therefore, this study will consider some factors to consider for comparison with overseas cases in estimating worker exposure dose.

To assess worker exposure doses, the classification of decommissioning activities must first be made. It should be classified including large components that can be generally considered, and the contents should be similar to compare with overseas cases. In case of decommissioned NPPs with prior experience, it is possible to predict worker's exposure with respect to plant capacity, but this does not seem to have a specific correlation when reviewing the related data. Depending on the plant capacity, the occurrence of contamination of radioactive materials may have some correlation, but it cannot be determined that it has causality with the worker's dose when dismantling. In addition, it is expected that the effects of workers' exposure doses will vary depending on when the highly contaminated SSCs will be dismantled from permanent shut down. Therefore, the decay correlation coefficient for this high radiation dose works should be considered. If the high radiation dose work is performed before the base year, a correlation coefficient larger than 1 value will be applied, and in the opposite case, a value less than 1 will be applied. Whether or not to perform Full System Decontamination (FSD) is also an important consideration that affects worker dose, and correlation factors should be applied.

In this study, the matters to be considered when estimating worker dose for dismantling NPPs were reviewed. This suggests factors to be reflected in the work classification and dose results for comparison with overseas NPP experiences. Therefore, when doing the workers' dose estimation, it is necessary to derive a normalized doses considering each correlation factor when comparing with overseas cases along with dose estimation for the dismantling activities.

Keywords: Decommissioning, Dose estimation, Dismantling activity, Radiation dose

Review of the Area Factor From Nuclear Facilities Decommissioned and Comparisons With Preliminary Results

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The Derived Concentration Guideline Level (DCGL) using RESRAD code is generally obtained for the reuse of the site and remaining buildings of the decommissioning of nuclear facilities. At this time, the evaluation first considers wide DCGL assuming homogenous contamination for the entire target site. The DCGL derived through this will be compared with the actual contamination measured at the Final Status Survey (FSS) stage to determine whether the site is compliance with criteria. Guidelines for Survey units are presented in MARSSIM and suggested in Class 1 through 3. Therefore, DCGL for the survey unit of a certain smaller area is established by applying a correction factor from wide DCGL, which is define as an Area Factor (AF). Therefore, this study reviewed the AF applied in overseas cases, reviewed the necessary factors for derivation, and compared them by applying factors to the preliminary experimental target area for domestic nuclear installations.

The AF is the ratio of the dose from the base-case contaminated area to the dose from a smaller contaminated area with the same radioactive concentration. To this end, an unrestricted resident farmer scenario was applied as the site reuse scenario, which deals with all exposure pathways considered in the RESRAD. The potential exposure pathways considered in resident farmer scenarios are largely divided into external and internal exposures, which are based on NUREG/CR-5512.

In addition, in order to calculate the AF, a change in the contaminated area occurs, and accordingly, a variable that varies according to the area, i.e., length parallel to aquifer flow (LCZPAQ), the contaminated fraction of plant food ingested (FPLANT), the contaminated fraction of meat and milk (FMEAT and FMILK), is accompanied. As the contamination area decreases, these variables decrease, and the criteria for reduction were reflected through overseas cases.

In this study, three nuclides (C-14, Co-60, and Cs-137) were assumed as representative nuclides, and the area of the contaminated site was selected as 50,000 m² and reduced at a certain rate. As a result, each nuclide showed different characteristics, but in general, AF increases as the area decreases. Compared to the area of this study, AF values were calculated to be smaller than those of overseas cases, but it was confirmed that the area of the values showed similar patterns. In addition, in the case of C-14, the slope of AF increased rapidly as the area decreased, while Co-60 and Cs-137 showed similar slopes.

Keywords: Decommissioning, Area factor, DCGL, Site release

Sensitivity Analysis of Site-Specific Parameters Considered for Site Reuse After the Decommissioning of Nuclear Facilities

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The effects of an individual effective dose from radioactive contamination that will remain during site reuse after the decommissioning of nuclear facilities is generally assessed using the RESRAD code. The calculated results should meet the site reuse criteria presented by regulators, 0.25 mSv/yr in the United States and 0.1 mSv/yr in Korea. After completion of decommissioning, the dose is not subject to measurement, resulting in Derived Concentration Guideline Level (DCGL) remaining at the site that is practically consistent with the dose criteria. In order to assess dose using the RESRAD code, various requirements will need to be considered and determined, where the selection of input parameters is one of the important factors in the dose assessment. In addition, appropriate selection of site-specific parameters is important to reflect the site characteristics of each decommissioned Nuclear Power Plant (NPP). Therefore, this study intends to analyze the impact of site-specific parameters by referring to the cases of overseas decommissioned NPPs.

In order to evaluate doses using RESRAD code, a site reuse scenario must first be selected. In general, in the case of unrestricted reuse, the resident farmer scenario can be applied, so the resident farmer scenario was also selected in this study. In addition, once a resident farmer scenario is selected, input parameters are selected according to the scenario, and the input parameter inputs a single value or distribution according to the deterministic or probabilistic evaluation method. Therefore, since this study is to evaluate the effect on site-specific parameters, a single value was applied as a deterministic evaluation method. For the 10 site-specific parameters considered in overseas cases, the difference was set twice using the F9 function key in the RESRAD code and the results were analyzed.

In this study, we used prior research data targeting domestic nuclear facility for sensitivity analysis. Related parameters include the category of contamination layer, soil, water transport, ingestion, and occupancy. The parameters that appeared as the greatest influence among the 10 parameters were different in radionuclide on the contaminated zone. We showed the changes according to the difference in input parameters was presented using the graph provided by the RESRAD code. As a result, in the evaluation for Co-60 in this study, no significant change was observed. However, in case of H-3, several parameters values were changed, indicating that the effect on dose will be different depending on the site characteristics of the nuclear facilities.

Keywords: Decommissioning, Sensitivity, DCGL, Site-specific parameter

Radiological Hazard Assessment for Remedial Actions in Contaminated Areas

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Radioactive materials deposited after nuclear accident or radiological emergency result in radiation exposure to individuals living in long-term contaminated territories. Therefore, the remedial actions should be taken on affected areas for the evacuated residents to return to their homes and normal lifestyle. Meanwhile, radiation exposure occurs through various pathways by work types during the site clean-up. Therefore, dose assessment is crucial to protect emergency workers and helpers from the potential radiological risk. This study estimated the exposure dose to individuals decontaminating the areas contaminated with ^{60}Co , ^{63}Ni , ^{90}Sr , ^{134}Cs , ^{137}Cs , and then calculated the maximum workable soil concentration to comply with the reference level of 20 mSv/y for transition to existing exposure situations. For the realistic assessment, the detailed exposure scenarios depending on the types of work (excavation, collection, transportation, disposal, landfill), and the relevant exposure pathways were used. In addition, with the LHS (Latin Hypercube Sampling) - PRCC (Partial Rank Correlation Coefficient) method, sensitivity analysis was performed to identify the influence of the input parameters and their variation on the model outcomes. As a result, the most severe exposure-induced type was identified as the excavator operation with an annual individual dose of 4.75E-01 mSv at the unit soil concentration (1 Bq/g), from which the derived maximum workable soil concentration was 4.21E+01 Bq/g. Dose contribution by isotopes were found to be ^{60}Co (55.63%), ^{134}Cs (32.01%), and ^{137}Cs (12.28%), and the impact of ^{63}Ni and ^{90}Sr were found to be negligible. Dose contribution by exposure pathways decreased in the following order: ground-shine, soil ingestion, dust inhalation, and skin contamination. Furthermore, the most high sensitive input parameters and their PRCC were found to be as the dilution factor (0.75) and as the exposure time (0.63). In conclusion, the results are expected to contribute to optimize radiation protection strategies for recovery workers and to establish appropriate response procedures to be applicable in areas with high deposition density after a radiological or nuclear emergency.

Keywords: Nuclear emergency, Site remediation, Radiation dose assessment, Sensitivity analysis, Radiation protection

Revitalization of Education for Radiological Emergency Preparedness

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Employees of nuclear licensees have to take the education for radiological emergency preparedness, as prescribed by presidential Decree. The Korea Atomic Energy Research Institute (KAERI), as an educational institution designated by the Nuclear Safety and Security Commission (NSSC), has been conducting field-oriented workplace education. This aims to enhance understanding of radiological emergencies that may occur in nuclear facilities and to strengthen response capabilities to prevent and deal with accidents in the event of radiation emergencies or radioactive disasters. To accomplish these educational goals, a paradigm shift from the previous theory-oriented curriculum to a participatory curriculum with high field applicability is needed to strengthen the ability to respond to nuclear or radiological emergencies. In addition, a feedback system is required to manage the quality of education and improve the curriculum. In this regard, KAERI sought ways to revitalize the education to strengthen the emergency response competencies. Based on the concept of the Systematic Approach to Training (SAT) methodology, which is recommended by the International Atomic Energy Agency (IAEA) for the development and implementation of education and training for NPP personnel, an educational model and its feedback system were developed. Then, a field-oriented participatory curriculum operation and satisfaction survey were conducted to evaluate the educational effectiveness. Lastly, the survey results were discussed in a critique session to point out weaknesses and indicate areas for improvement, and then were used as data for educational quality assurance. This paper introduces the composition and effectiveness of KAERI's SAT-based education model based on its recent three years of experience.

Keywords: Nuclear or radiological emergencies, Emergency preparedness, Emergency response, Education for radiological emergency response

Development of Boat Sampling Technology for Defects Analysis of Welded Parts of Bottom Head Penetration Nozzles of Reactor Vessel

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In the pressurized water nuclear reactors (PWRs), the upper and bottom head penetration nozzles, the geometric asymmetry of the welded part increases from the center to the outer part, increasing the possibility of defects. For this reason, it is important to perform early detection and management through analysis of defects occurring in the welded parts of upper and bottom penetration nozzles of reactor vessel. However, it is very difficult to operate boat sampling of the welding area because the spacing of the penetration nozzle of the bottom head of the reactor is very narrow. In addition, it is more difficult to collect welded specimens of bottom penetration nozzles by electrical discharge machining in a boric acid water environment of nuclear reactor. In this work, to overcoming these technical difficulties, we developed a boat sampling robot system, which is composed of the specimen collection electrode head, borate-mediated discharge electrode and control system. Also, we performed basic performance tests and summarize the results.

Acknowledgements

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Keywords: Boat sampling, Electrical discharge machining (EDM), Bottom head penetration nozzle, Borate-mediated discharge electrode

Evaluation of Internal Exposure Based on the Consumption of Food Produced on the Site After Decommissioning

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The site used for a nuclear facility can be released after decommissioning if the results of dose estimation meet the regulatory requirements and the site release is approved by the regulatory body. RESRAD-ONSITE, developed by the Argonne National Laboratory, is a computer code used to estimate the dose to the residents on radiologically contaminated sites. The dose estimation for site release should consider various exposure pathways, including inhalation, ingestion, and external exposure. This study used RESRAD-ONSITE to evaluate the internal exposure dose and identify radionuclides due to the intake of food produced on radiologically contaminated sites.

The upper limit of the clearance level of radionuclides expected to remain at the site was used as the source terms for the dose evaluation. In addition, the amount of food intake per capita was obtained from eight countries using nuclear power generation as of 2020. The default values of RESRAD were used for other parameters except for intake by type of food and source terms.

As a result of the dose evaluation, the contaminated water and vegetables showed a great contribution to the exposure dose. The dose due to tritium in drinking water was highest in the third year. In addition, regarding the intake of vegetables, the internal exposure due to ^{90}Sr was the highest in the first year.

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Keywords : RESRAD-ONSITE, Internal exposure, Decommissioning, Dose estimation

Evaluation of the Potential for Worker Exposure During Maintenance Operations in the Phosphate Industry

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The phosphate industry is classified by IAEA as one of the Naturally Occurring Radioactive Materials (NORM) industry sectors most likely to require regulatory consideration. The production of phosphorous fertilizers constitutes the major activity in the industry, which can give rise to exposures of workers and the public through the handling and usage of phosphate rock and residues associated with processing. During the production process, when phosphate rock is digested with acid to produce phosphoric acid, some radionuclides, particularly Radium, become concentrated in residues, such as the scale that tends to form inside pipes and vessels.

The registered radioactivity of phosphate rock in South Korea is less than 1.7 Bq/g for U-238, but according to the IAEA SRS No. 49, the radioactivity of phosphate scale can be up to 1,000 times higher than the raw mineral. Therefore, this study evaluated the potential for worker exposure during maintenance related to the removal of scales at a fertilizer manufacturing facility producing phosphoric acid in Korea.

Acknowledgements

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Keywords: NORM residues, Phosphate industry, Worker exposure, Graded approach

Review on Radiological Environmental Impact Assessment Regulatory Trend

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In 2022 and 2023, the Korea Institute of Nuclear Safety (KINS), a regulatory body, revised the regulatory guidelines for off-site dose evaluation to residents, marine characteristics surveys around nuclear facilities, and environmental radiation surveys and evaluation around nuclear facilities. In addition, the NRC, a US regulatory body, has revised regulatory guide 1.21 (MEASURING, EVALUATING, AND REPORTING RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS AND SOLID WASTE) to change environmental programs for nuclear facilities.

The domestic regulatory guidelines were revised and added to reflect the experience of site dose evaluation for multiple units during the operation license review of nuclear facilities, the resident exposure dose age group was modified to conform to ICRP-72, and the environmental monitoring plan was clarified. In the case of the US, the recommended guidelines for updating the long-term average atmospheric diffusion factor and deposition factor, the clarification of the I-131 environmental monitoring guidelines for drinking water, and the clarification of the procedures described in the technical guidelines when changing environmental programs have been revised and added. Through such regulatory trend review, it is necessary to preemptively respond to changes in the regulatory environment in the future.

Keywords: RER, NRC REG, GUIDE 1.21, ODCM

Study on Improvement of Gaseous Effluent Evaluation of Radiological Environmental Impact Assessment

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K-DOSE60, a off-site dose calculation program currently used by khnp, is performing evaluation based on the gaseous effluent evaluation methodology of NRC Reg. Guide 1.109. In particular, H-3 and C-14, which are the major nuclides of gaseous effluent, are evaluated using a ratio activity model. Among them, H-3 is additionally evaluating the dose to OBT (Organically Bound Tritium) and HT as well as HTO (Triated water). However, NRC Reg. Guide 1.109 is a methodology developed in the 1970s, and verification was performed by applying the evaluation methodology of H-3 and C-13 presented by IAEA TRS-472 in 2010 to the current K-DOSE60. The IAEA TRS-472 methodology also includes OBT and HT for H-3.

In order to apply the ratio radioactivity model presented in IAEA TRS-472, the absolute and relative humidity were calculated using the weather tower of the nuclear site and used for H-3 evaluation. For the dose evaluation of HT, the previously used Canada Chalk River Lab. (CNL) conversion factor was used. For atmospheric carbon concentration, the carbon concentration presented in IAEA TRS-472 was used, not the carbon concentration in the 1970s of NRC Reg. Guide 1.109. It was confirmed that the K-DOSE60, which applied the changed input data and methodology, was satisfied by performing comparative verification with the numerical calculation value.

Keywords: RER, GASEUOS EFFLUENT, ODCM, IAEA TRS-472

Review on Radiological Environmental Impact Assessment Technology Trends in Case of Accidents

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After the Fukushima nuclear power plant accident in 2011, interest in technology for evaluating residents' exposure to effluents generated from nuclear power plants at the time of the accident has increased. KHNP has developed the S-REDAP program and is using it to evaluate radiation dose and recommend resident protection measures in the event of a nuclear power plant emergency. Its main functions are source term evaluation, atmospheric diffusion evaluation, radiation dose evaluation, etc. Based on these evaluations, resident protection measures are evaluated.

In Japan, evaluation is conducted through a program called SPEEDI-MP (System for Prediction of Environmental Emergency Dose Information Multi-model Package) created by JAEA (Japan Atomic Energy Agency). Similar to S-REDAP, the program also evaluates effluents emitted from nuclear facilities through source term evaluation and atmospheric diffusion factor evaluation. In JAEA, through a program using SPEEDI-MP, the source term evaluation was performed in collaboration with NSC (Nuclear Safety Commission) in the event of the Fukushima nuclear plant accident, and dose evaluation in Japan was performed 2 months as an atmospheric diffusion factor using meteorological data for 2 days. Through comparative analysis of evaluation data from Japan, improvements to the current program be derived.

Keywords: S-REDAP, SPEEDI-MP, RER

Review on Overseas Radiological Environmental Impact Assessment Program Development Status

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The US NRC developed a program called NRCDOSE3 to evaluate the environmental impact of radiation around nuclear facilities. The NRCDOSE3 code is a software suite that integrates the functionality of three individual LADTAP II, GASPAR II, and XOQDOQ Fortran codes that were developed by the NRC in the 1980's and have been in use by the nuclear industry and the NRC staff for assessments of liquid effluent and gaseous effluent, and meteorological transport and dispersion, respectively. Through the integrated program, it is possible to conduct safety assessment and environmental impact assessment from liquid and gaseous effluent when operating permits are granted. In addition to a more user-friendly graphic user interface (GUI) for inputting data, significant changes have been made to the data management and operation to support expanded capabilities.

The basic calculation methods of the LADTAP II, GASPAR II, and XOQDOQ have not been changed with this update to the NRCDOSE3 code. Several features have been added. The previous program used only ICRP-2 dose conversion factor, but the new program can additionally use dose conversion factor of ICRP-30 and ICRP-72. In the previous program, 4 age groups (infant, child, teen, and adult) were evaluated during dose evaluation, but when ICRP-72 was selected, 6 age groups (infant, 1-year, 5-year, 10-year, 15-year, and adult) could be evaluated. In addition, when selecting ICRP-72, many user-modifiable parameters such as food intake and exposure time were added. It will be referred to E-DOSE60, a program currently under development.

Keywords: RER, ODCM, NRCDOSE3, E-DOSE60

Performance Evaluation of NaI(Tl) Scintillator Detector for Gamma Monitoring System

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In-situ gamma spectrometer with mobile equipment can be used for rapid determination of radioactivity in the environment within a very short interval. 2”×2” NaI(Tl) scintillator are used to build a mobile radiation measurement system (called as Monitoring of Ambient Radiations of KAERI for Backpack, MARK-B3) with a signal processing unit, and GPS and interface units to a PC for wireless controlling system. Development of the survey system is to measure ambient gamma-ray spectrometry for estimating ground radioactivity and radiation dose in the environment. The ambient dose rate is estimated using G-factor method.

For determination of G-factor, we conducted MCNP simulations in assumptions of various incident photons into the detector system. And the scintillator was exposed to Cs-137 source in the range of 1-300 mGy/hr. Calculated dose rates for different simulation results were compared to the irradiated dose rate to derive correction factor of G-factor. To evaluate performance of the MARK-B3, in-situ gamma spectrometry was conducted in Jeju island.

Keywords: In-situ gamma spectrometry, Dose rate spectrometry, Halide scintillator, Environmental radiation survey

Comparison of Uranium Analysis Methods Using Alpha Spectrometer and Thermal Ionization Mass Spectrometer (TIMS)

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Radionuclide analysis methods must be secured in the event of emergencies such as the discovery of unknown nuclear material or nuclear accidents in neighboring countries or Korea. Most institutions in Korea are in their early stages of radionuclide analysis method development and do not even have Radiation Controlled Areas where they can handle the samples safely. Some institutions such as the Korea Atomic Energy Research Institute have the ability to perform radionuclide analysis for nuclear facilities or verification of nuclear activities. In Korea, it is necessary to secure nuclide analysis technology to enable independent verification in times of emergency or need. This paper analyzes uranium as the target nuclide using alpha spectrometer and TIMS. Alpha spectrometer detects alpha particles emitted from uranium samples and measures the concentration of uranium isotopes. This method has a high selectivity that distinguishes it from other elements, and accurate measurements can be made even when uranium samples are mixed with other elements. In addition, there is minimal interference from other radioactive isotopes in the sample, and the sample preparation is simple, resulting in relatively short analysis times. In contrast, TIMS detects ionized uranium ions by heating the uranium sample. This method may have potential interference from other elements and may take relatively longer analysis times. However, TIMS has high sensitivity and accuracy and can detect various elements other than uranium, making it suitable for various analyses. Therefore, when analyzing uranium, it is recommended to select and use the appropriate device according to the purpose, as both alpha spectrometer and TIMS have their pros and cons. Furthermore, by using both devices in parallel, more accurate and reliable results can be obtained. This paper aims to compare the analysis methods of alpha spectrometer and thermal ionization mass spectrometry, which are widely used for nuclide analysis in unknown nuclear materials.

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Keywords: Alpha spectrometer, TIMS, Uranium, Nuclide analysis

Determination of the Inherent Filtration of the 320 kV X-ray Tube at KHNP-CRI by HVL Measurement

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According to ISO 4037, the thickness of the inherent filtration for the radiation qualities L-40 to L-240, N-40 to N-400, W-60 to W-300 and H-80 to H-400 shall be equivalent to 4 mm Al for matched reference radiation fields or adjusted as far as possible to 4 mm Al for characterized reference radiation fields. And for matched reference fields, the tube window must be made of beryllium and its thickness should not exceed 10 mm. In the case of characterized reference fields, the thickness of the beryllium window should not exceed 10 mm, but it is acceptable to use an aluminum window with a maximum thickness of 1.5 mm. 320 KV X-ray tube installed at KHNP-CRI has been designed to equipped with a 3 mm Be for tube window and an additional 4 mm Al to obtain a total inherent filtration equivalent to that of 4 mm Al. In the previous study, the inherent filtration of 320 kV X-ray tube at KHNP-CRI has been verified by MCNP simulation. However, the ISO standards suggest a method for determining the thickness of the inherent filtration by half-value layer (HVL) measurement and spectrometry. In this regard, the inherent filtration was reassessed using HVL measurement.

To determine the inherent filtration, 1st HVL of the beam generated by the tube at a tube potential 60 kV was measured. The measurements were conducted with a calibrated spherical ionization chamber (model A3, Exradine) placed at a distance of 1 m from the target, at the center of the radiation field size. The X-ray tube current was set to 2 mA. The thickness of aluminum absorbers was gradually adjusted in subsequent measurements until approached the 1st HVL. 1st HVL were estimated using the linear regression equation computed with the current values for the thickness of the absorbers. As a results, the thickness of the 1st HVL was estimated as 2.845 mm Al. According to the correlation between the inherent filtration and 1st HVL suggested in ISO standard, the value of the inherent filtration was deduced as 4.25 mm Al that is rounded to the nearest 0.05 mm by interpolation. Further studies on the effects of the inherent filtration thickness determined in this study will be conducted.

Keywords: X-ray, ISO 4037, Inherent filtration, Half-value layer

Overview of Tritium Criteria in Drinking Water

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Tritium is a radioactive isotope of hydrogen with a half-life of about 12.3 years, and it is commonly found in the environment as a result of the production of Nuclear Power Plants. The World Health Organization (WHO) has established guidelines for the permissible levels of tritium in drinking water. The guideline value for tritium in drinking water is 10,000 Bq/L.

It is important to note that the guideline value for tritium is not a legal limit, but rather a recommendation. National and local authorities may establish legal limits that are more restrictive than the WHO guideline value based on local conditions and risk assessments.

The Australia and Finland have set a limit for tritium in drinking water at 76,103 Bq/L and 30,000 Bq/L respectively, which is more than three to seven times higher compare to guideline value of WHO. The United States Environmental Protection Agency (EPA) has set a maximum contaminant level (MCL) for tritium in drinking water at 20,000 picocuries per liter (pCi/L), which is equivalent to 740 Bq/L. The Health Canada has set a guideline value for tritium in drinking water at 7,000 Bq/L. Assuming drinking water corresponding to each tritium limit (or guideline value) for one year, the expected exposure dose is 0.01 mSv to 1 mSv. It means that the tritium in drinking water below the limits or guideline value does not pose a significant risk to human health.

Keywords: Tritium, Drinking water, Criteria, Dose exposure

Limits of Dose Management for Transport Worker in Nuclear Power Plant

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According to attached Table 1 of the Enforcement Ordinance of the Nuclear Safety Act, the effective dose limit of transport workers shall not exceed 6 mSv per year. In addition, the enforcement ordinance defines a transport worker as a person who transports radioactive substances outside the radiation management area and does not correspond to a radiation worker.

In the nuclear power plants (NPPs), substances in radiation management areas are frequently transported inside or outside the plant. During loading of substances in the radiation management area onto the vehicle, the transport workers (including driver) are located outside the radiation management area. And also the exposure dose of transport workers is managed by using Automatic Dose Reader (ADR). However, the exposure dose of transport workers managed by NPP licensee is limited to the exposure caused by the transport actions required by the plant. This means that radiation exposure caused by the transport of radioactive materials carried out separately by individual transport workers other than the plant requirements cannot be managed. Therefore, even if the NPP licensee manages the transport worker's dose below 6 mSv, it is difficult to guarantee that the total annual exposure dose, including the transport worker's individual transport behavior, is less than 6 mSv. Therefore, it would be appropriate to manage the dose of the transport worker by the transport worker's agency rather than by the NPP licensee.

Keywords: Delivery worker, Radiation exposure, Dose limits, Dose management

Progress in Improvement of Safety Information Transmission System

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In response to a regulatory mandate, all nuclear licensees are obligated to establish an information system that can provide essential information in the event of a radiation emergency by connecting the monitoring data of the Safety Parameter Display System (SPDS) or equivalent system to the Korea Institute of Nuclear Safety (KINS). Responding to this responsibility, the Korea Atomic Energy Research Institute (KAERI) has established the Safety Information Transmission System (SITS), which enables the collection and real-time monitoring of safety information.

The KAERI monitors and collects safety information, which includes data from the HANARO Operation Work Station (OWS) and the HANARO & HANARO Fuel Fabrication Plant (HFFP) Radioactivity Monitoring System (RMS), and the Environmental Radiation Monitoring System (ERMS) & meteorological data. Currently, the transmission of this safety information to the AtomCARE server of the KINS takes place via the SITS server located in the Emergency Operations Facility (EOF). However, the multi-path of transmission through SITS has caused problems such as an increase in data transmission interruptions and errors, as well as delays in identifying the cause and implementing system recovery measures. To address these issues, a new VPN is currently being constructed on the servers of nuclear facilities that generate and manage safety information to establish a direct transmission system of safety information from each nuclear facility to the AtomCARE server. The establishment of a direct transmission system that eliminates unnecessary transit steps is expected to result in stable information transmission and minimize the frequency of data transmission interruptions.

As of the improvement progress, a security review was conducted in the second and third quarters of 2022 to evaluate the security of newly introduced VPNs to the nuclear facility server, and based on the results of the review, security measures were strengthened. In the fourth quarter of 2022, the development of a direct transmission system for safety information began, and it is scheduled to be completed by the fourth quarter of 2023. The project includes the construction of the transmission system, system inspection, and comprehensive data stability testing. Afterward, the existing SITS located in the EOF will be renamed as the Safety Information Display System (SIDS), and there are plans to remove any unused servers and VPNs.

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Keywords: Nuclear emergency preparedness, Safety information, Safety information transmission system (SITS), VPN

Assessment of Dose Rate and Radioactivity Ratio in Sites Around Nuclear Power Plants

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In this research, the dose rate was measured using a backpack-type scan survey device at 4 sites in sites around Nuclear Power Plants (Kori, Wolsong, Hanbit, Hanul), and the radioactivity ratio for each nuclide was evaluated using an high-purity germanium (HPGe) detector. Kori, Wolsong and Hanul power plants were measured within 2 km of the power plant, and Hanbit power plants were measured about 6.7 km from the power plant. As a result of measuring the dose rate with a backpack-type scan survey device, the average dose rate was the lowest in the measurement site 1 at 0.090 $\mu\text{Sv/h}$, and the highest in the measurement site 4 at 0.145 $\mu\text{Sv/h}$. All measurement points showed the domestic environmental dose rate level. The data obtained by the scan survey was visualized using the classed post and gridding functions of the surfer program. As a result of measurement with the HPGe detector, ^{137}Cs was not detected, and only natural nuclides were detected. Among the detected natural nuclides, the radioactivity ratio was the highest for ^{40}K with an average of 94.56%, and the lowest for ^{214}Pb with an average of 0.26%. The results of this research can be used as basic data for radiation environment surveys around nuclear power plants. Further studies are needed to evaluate the radiation impacts by region and environment through periodic measurements.

Keywords: Dose rate, Visualization, Radioactivity ratio, Radiation environment survey

Current Status of Occupational Exposure in Korean Nuclear Power Plants

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The 2007 Recommendation of the International Commission on Radiological Protection recommended the application of dose constraints to optimize radiation protection to resolve the inequity of exposure among radiation workers. The average annual occupational doses in Korean nuclear power plants (NPPs) are 0.3-0.8 mSv. These doses are much lower than the annual effective dose limit of 50 mSv for radiation workers stipulated by the Nuclear Safety Act. In addition, most NPP workers received less than 0.1 mSv per year. These doses are lower than the average annual occupational doses of 0.3-0.8 mSv. Korean regulatory body conducted the study to legislate the dose constraints in the Korean regulatory system and determine dose constraints (draft) for radiation workers. The legislation of dose constraints would not greatly affect the radiation protection programs in Korean NPPs because most workers received very low doses. However, some workers received relatively higher doses than others.

This study analyzed the occupational exposure conditions, such as exposure type and situation, in Korean NPPs. This study investigated the internal and external radiation doses and the radiation doses depending on the NPP operating conditions, including normal operation, planned maintenance, and intermediate maintenance, for the last ten years (2012-2021). As a result, most NPP workers received external exposure rather than internal exposure. Furthermore, most radiation exposures occurred during the planned maintenance period. The results of this study can be used for optimizing occupational doses in Korean NPPs.

Acknowledgements

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Keywords: Occupational exposure, Nuclear power plants (NPPs), Dose constraints, Dose distribution

7분과

방사화학(Oral)

A Study on the Determination of Organic Chelating Agents in a Variety of Radioactive Wastes

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Low- and intermediate-level radioactive waste for permanent disposal often contains organic complexing agents, so-called chelating agents. Organic complexing agents, which are polycarboxylic acids, can increase the mobility of radionuclides into the environment by forming water-soluble complexes with most heavy metals. Therefore, analyzing the complexing agents in radioactive waste is crucial for comprehensive management of nuclear wastes. According to regulatory guidelines, specifically Notice No. 2021-16 issued by the Nuclear Safety and Security Commission, the determination of chelating agent content in radioactive waste materials is required to ensure proper management and safe disposal. However, only a few methods are available to analyze the chelators in various matrices such as concrete, metals, soil, and mixed solid wastes like plastics, vinyl, and rubber. Recently, we found a UV-Vis method based on an enzymatic reaction is inadequate for analyzing citric acid in radioactive waste with a complex matrix like concrete. To address this, we developed a method to determine the contents of EDTA and NTA using a UV-Vis spectrophotometer and citric acid using ion chromatography. The results showed good validity and reliability to determine the chelating agents in various radioactive wastes.

Keywords: Organic complexing agents, Chelating agents, Ion chromatography, Enzymatic method, UV-Vis spectrophotometer

Potential of Indigenous Bacteria Driven U(VI) Reduction Under Anaerobic Condition

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Bacterial metabolisms influence the behavior of uranium (U) in deep geological repository (DGR) system because bacteria are ubiquitous in the natural environment. Nevertheless, most studies for the U(VI) bioreduction have focused on a few model bacterium, such as *Shewanella putrefaciens*, *Desulfovibrio desulfuricans*, and *Geobacter sulfurreducens*. In this study, the potential of aqueous U(VI) ($U(VI)_{aq}$) reduction by indigenous bacteria was examined under anaerobic conditions with addition of 20 mM sodium acetate for 24 weeks. Three different indigenous bacterial communities obtained from granitic groundwater at depths of 44–60 m (S1), 92–116 m (S2), and 234–244 m (S3) were applied for $U(VI)_{aq}$ reduction experiments. The S2 groundwater contained the highest U concentration of 885.4 $\mu\text{g/L}$ among three groundwater samples, where U mainly existed in the form of $\text{Ca}_2\text{UO}_2(\text{CO}_3)_3(\text{aq})$. The S2 groundwater amended 20 mM of sodium acetate was used for the $U(VI)_{aq}$ bioreduction experiment. Variations in the $U(VI)_{aq}$ concentration and redox potential were monitored for 24 weeks to compare $U(VI)_{aq}$ removal efficiency in response to indigenous bacteria. The $U(VI)_{aq}$ removal efficiencies varied among three indigenous bacteria: 57.8% (S3), 43.1% (S2), and 37.7% (S1). The presence of the thermodynamically stable uranyl carbonate complex resulted in the incomplete $U(VI)_{aq}$ removal. Significant shifts in indigenous bacterial communities were observed through high-throughput 16S rRNA gene sequencing analysis. Two SRB species, *Thermodesulfovibrio yellowstonii* and *Desulfatirhabdium butyrativorans*, were dominant in the S3 sample after the anaerobic reaction, which enhanced the bioreduction of $U(VI)_{aq}$. The precipitates produced by bacterial activity were determined to be U(IV)-silicate nanoparticles by a transmission electron microscope (TEM)-energy dispersive spectroscopy (EDS) analysis. These results demonstrated that considerable U immobilization is possible by stimulating the activity of indigenous bacteria in the DGR environment.

Keywords: Uranium, Bioreduction, Deep geological repository, Sulfate-reducing bacteria

Dissolution Behaviors of ThO₂ and PuO₂ in the Synthetic Groundwater Contacted With Air

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Dissolution behaviors of ThO₂(cr) and PuO₂(cr) in synthetic groundwater were investigated at room temperature ($23 \pm 2^\circ\text{C}$) under atmospheric conditions. The synthetic groundwater was prepared according to the chemical composition of the KURT-DB3 groundwater. The pH and E_h of the synthetic groundwater were pH 8.9 and 0.5 V, respectively, and the major components were Na, K, Ca, Mg, Si, Cl, SO₄, F and HCO₃ ions. A few mg of ThO₂(cr) and PuO₂(cr) powder were added in the synthetic groundwater and the concentrations of Th and Pu in supernatant were monitored for 5 months of reaction time. The concentrations of Th before and after ultracentrifugation were compared, while the solid-liquid phase separation of Pu samples could not be applied due to the small volume of sample solutions. The concentrations of Th and Pu were measured by ICP-MS and alpha spectrometry, respectively. Geochemist's Work Bench (GWB, standard, 17.0) was applied for the modeling with ThermoChimie TDB v. 11a, which was updated with the latest NEA-TDB (vol. 14). Aqueous species distributions and solubility limiting solid phases of Th and Pu under the synthetic groundwater conditions were evaluated. The results of geochemical modeling indicate that aqueous Th-OH-CO₃ ternary species and Pu(IV) species are dominant in solutions equilibrated with ThO₂(s) and PuO₂(am, hyd), respectively. The dissolution behaviors of ThO₂(cr) and PuO₂(cr) are comparable to the dissolution of ThO₂(aged, $\log K_{sp} = 8.5$) and the oxidative dissolution of PuO₂(am, hyd) in the presence of PuO₂(coll, hyd), respectively.

Keywords: Thorium, Plutonium, Dissolution, Synthetic groundwater, KURT-DB3

Introduction to Chemical and Electrochemical Measurement Methods of Oxide Ion Concentration in Molten Salts

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Measurement of oxide ion (O^{2-}) concentration is a basic technology required in molten salt fields, from energy storage systems to electrolytic reduction of rare earth elements or spent nuclear fuels. In a molten salt reactor, O^{2-} ions react with actinide elements to form their oxides or oxy-chlorides to induce actinide precipitation, and promote metal corrosion to cause a failure of structural material. For these reasons, removal of O^{2-} ions and monitoring of the O^{2-} concentration in molten salt reactors are essential.

In this study, methods using chemical and electrochemical methods were investigated for measuring the concentration of O^{2-} ions in a molten salts. The acid-base neutralization reaction was used as a chemical analysis method. And electrochemical methods using the O^{2-} diffusion limit current and YSZ (yttria stabilized zirconia) indicator electrode were used for measuring the O^{2-} concentration. Finally, a modified method using porous membrane electrode was applied to monitor the O^{2-} concentration. The O^{2-} concentration was measured up to about 2wt% of Li_2O by the method using the O^{2-} diffusion current, up to about 4wt% by the YSZ indicator electrode, and about 6wt% by the porous membrane electrode in $LiCl$ molten salts.

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Keywords: Electrochemical measurement, Oxide Ion (O^{2-}) concentration, Molten salts

Corrosion Mitigation Strategies for Ni-based Alloy in NaCl-MgCl₂ Melts at 973 K

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A molten salt reactor (MSR) is a conceptual nuclear reactor that uses molten salt with liquid fuel as its primary coolant. Based on the thermophysical and neutronic properties, MSR has advantages such as high efficiency, safety, combustion of transuranic (TRU) elements, and availability of miniaturization and on-power refueling. Various research on MSR such as system development, neutronic analysis, material development, and molten salt property analysis has been conducted, but the biggest problem is the molten salt corrosion.

The molten salt corrosion on structural materials can be explained by two processes; electrochemical and chemical reactions. The reduction of oxidative ions such as fuel and TRU elements is one of the major causes of molten salt corrosion. Contamination by humidity and oxygen is also known as the accelerating factor of molten salt corrosion. Also, molten salt corrosion behaviors on structural material deteriorate when dissimilar alloys are introduced in the molten salt system. Various techniques to mitigate molten salt corrosion in fluoride system has been developed, but these are not well-verified in chloride system.

In this research, various methodologies to mitigate molten salt corrosion are studied. The corrosion behaviors of 80Ni-20Cr alloy in molten eutectic NaCl-MgCl₂ salt at 973 K are analyzed with various applications such as salt purification, sacrificial metal injection, and salt redox potential control. Oxygen and water impurities that can accelerate molten salt corrosion have been removed by electrochemical and chemical methods; Applying the reduction potential for H⁺/H₂ and oxidation potential for O²⁻/O₂, introducing HCl and CCl₄ gas, and introducing the metallic Cr and recovering the ionized Cr. Corrosion acceleration/deceleration effects were analyzed when introducing the reducing reagent such as Mg and Nb or oxidizing reagent such as metallic Mo and the effect of inert metallic element (W) was also investigated. The salt potential was controlled by applying the potential to the salt and adjusting the Eu³⁺/Eu²⁺ ratio.

Acknowledgements

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Keywords: Molten salt corrosion, Ni-based alloy, Redox potential control, Salt purification, Sacrificial metal addition

Development of a Machine Learning Model for Determination of Electrode Surface Area With Voltammetry of Two Rod Electrode in Molten Salts

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Radioactive wastes, including used nuclear fuel and decommissioning wastes, have been treated using molten salts. Electrochemical sensors are one of the options for in-situ process monitoring using molten salts. However, in order to use electrochemical sensors in molten salt, the surface area must be known. This is because the surface area affects the current of the electrode. Previous studies have used a variety of methods to determine the electrode surface area in molten salts. One method of calculating the electrode surface area is to use the reduction current peak difference between electrodes with known length differences. The method is based on the reduction peak and has the benefit of providing long-term in-situ monitoring of surfaces immersed in molten salt. A number of assumptions have been made regarding this method, including that there is no mass transport by migration or convection; the reaction is reversible and limited by diffusion; the chemical activity of the deposit should be unity; and species should follow linear diffusion. For the purpose of overcoming these limitations, a variety of machine learning algorithms were applied to different voltammogram datasets in order to calculate the surface area. Voltammogram datasets were collected from multiarray electrodes, comprising a multiarray holder, two tungsten rods (1 mm diameter) working electrodes, a quasi-reference electrode, and a counter electrode. The multiarray electrode holder was connected to the auto vertical translator, which uses a servo motor, for changing the height of the rod in the molten salts. To make big and diverse data for training machine learning models, various concentrations of corrosion products (Cr, Fe) and fission products (Eu, Sm) in NaCl-MgCl₂ eutectic salts were used as electrolyte; electrolyte temperatures were 500, 525, 550, 575, and 600°C. This study will demonstrate the potential of utilizing machine learning based electrochemical in situ monitoring in molten salt processing.

Acknowledgements

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Keywords: Pyroprocessing, Electrochemical monitoring, Artificial intelligence, Cyclic voltammogram, Multiarray electrode

Current Status and Development of Thermal Physical Properties Database for Molten Chloride Salts

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Molten salts have gained significant attention as a potential medium for heat transfer or energy storage and as liquid nuclear fuel, owing to their superior thermal properties. Various fluoride- and chloride-based salts are being explored as potential liquid fuels for several types of molten salt reactors (MSRs). Among these, chloride-based salts have recently received attention in MSR development due to their high solubility in actinides, which has the potential to increase fuel burnup and reduce nuclear waste production. Accurate knowledge of the thermal physical properties of molten salts, such as density, viscosity, thermal conductivity, and heat capacity, is critical for the design, licensing, and operation of MSRs. Various experimental techniques have been used to determine the thermal properties of molten salts, and more recently, computational methods such as molecular dynamics simulations have also been utilized to predict these properties. However, information on the thermal physical properties of salts containing actinides is still limited and unreliable. In this study, we analyzed the available thermal physical property database of chloride salts to develop accurate models and simulations that can predict the behavior of molten salts under various operating conditions. Furthermore, we conducted experiments to improve our understanding of the behavior of molten salts. The results of this study are expected to contribute to the development of safer and more efficient MSRs.

Keywords: Molten salt, Chlorides, Thermal physical property, Density, Viscosity

7분과

방사화학(Poster)

Surface Chemistry of Aquatic Strontium Adsorption Onto Synthesized Sodium Titanates

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The operation of nuclear power plants, nuclear waste depositories, and the decontamination and decommissioning of nuclear power plants all have the possibility of generating various kinds of radionuclides that can be formed as gaseous or liquid phases. Among the radionuclides, strontium is considered as most harmful substance due to its abundance in nuclear accident effluent, long half-life, high fission yield, high water solubility, and high mobility in aquatic environment. To remove strontium from aquatic environment, adsorption technique is mainly used with high economic feasibility, efficiency, and selectivity. Previously, we synthesized sodium titanates with mid-temperature hydrothermal method as selective strontium adsorbent in aqueous solution. Moreover, it was demonstrated that synthesized sodium titanates show high strontium adsorption rate with high selectivity with high surface area, pore diameter and volume.

Herein, we investigated the surface structure of synthesized sodium titanates before and after strontium adsorption in aqueous solution using scanning electron microscopy (SEM), energy dispersive X-ray spectroscopy (EDS), and X-ray photoelectron spectroscopy (XPS) analysis. According to SEM and EDS experimental results, aquatic strontium can be adsorbed as surface precipitation with formation of cube-shaped structure, which is quite similar strontium titanate structure crystals onto the surface of sodium titanates. In addition, XPS experimental results revealed that the titanium ions on the surface of sodium titanates were oxidized during strontium surface precipitation process, and the sodium ion on the surface of sodium titanates were exchanged with aquatic strontium ions via ion exchange process during strontium adsorption process.

Acknowledgements

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Keywords: Strontium, Adsorption, Surface chemistry, Sodium titanate

Chemistry and Design Considerations for Zinc Application Into Reactor Coolants of I-SMR

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Korean innovative SMR has been implemented developing with improved safety/economy and i-SMR technology development project to secure a competitive edge in SMR. For nuclear power plants, according to the revision of the Nuclear Safety Act (2013.6), it is mandatory to be reflected in the aging management program of nuclear power plants, and the aging management and regulation of major nuclear power plants are being strengthened. For i-SMR, chemistry environment and management strategy is essential to mitigate corrosion and radiation fields, since it has compacted and integrated module designs.

Since 1994, zinc injection into the reactor coolant system (RCS) has been applied more than 100 PWRs in the world to mitigate primary water stress corrosion cracking (PWSCC) and to reduce out-of-core radiation fields. In domestic NPPs, 7 have been applying zinc injection and had up to 90% radiation field reductions. For this reason, SMR needs to apply zinc injection for chemistry strategy.

Zinc target concentration will be 5~40 ppb at i-SMR, based on Ni-Fe-Cr materials as same as PWRs. Zinc injection location is in volume and purification control system between the volume control tank and charging P/P where the pressure is moderate. Zinc injection skid can consist of two micro-controllable pump (one for operation and one for stand-by) and one injection tank (batching tank for zinc solution). Zn, Ni, Si, Fe, and activated corrosion products should be monitored to identify zinc injection controls and trends. Flux mapping for core performance monitoring should be evaluated.

The application of zinc will be essential and effective and bring sustainable reliability for corrosion control and mitigation strategy to meet the risk-free i-SMR development.

Keywords: Zinc injection, Primary water stress corrosion cracking (PWSCC), Reactor coolant system (RCS), Radiation fields

Experimental Analysis and Mathematical Prediction of Radionuclides' Solubility Using Support Vector Machine Model: Effect of Organic Complexing Agents on Radionuclides' Solubility

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The organic complexing agents such as ethylenediaminetetraacetic acid (EDTA), nitrilotriacetic acid (NTA), and isosaccharinic acid (ISA) can enhance the radionuclides' solubility and have the potential to induce the acceleration of radionuclides' mobility to a far-field from the radioactive waste repository. Hence, it is essential to evaluate the effect of organic complexing agents on radionuclide solubility through experimental analysis under similar conditions to those at the radioactive waste disposal site. In this study, five radionuclides (cesium, cobalt, strontium, iodine, and uranium) and three organic complexing agents (EDTA, NTA, and ISA) were selected as model substances. To simulate environmental conditions, the groundwater was collected near the repository and applied for solubility experiments. The solubility experiments were carried out under various ranges of pHs (7, 9, 11, and 13), temperatures (10°C, 20°C, and 40°C), and concentrations of organic complexing agents (0, 10^{-5} , 10^{-4} , 10^{-3} , and 10^{-2} M). Experimental results showed that the presence of organic complexing agents significantly increased the solubility of the radionuclides. Cobalt and strontium had high solubility enhancement factors, even at low concentrations of organic complexing agents. We also developed a support vector machine (SVM) model using some of the experimental data and validated it using the rest of the solubility data. The root mean square error (RMSE) in the training and validation sets was 0.012 and 0.016, respectively. The SVM model allowed us to estimate the solubility value under untested conditions (e.g., pH 12, temperature 30°C, ISA 5×10^{-4} M). Therefore, our experimental solubility data and the SVM model can be used to predict radionuclide solubility and solubility enhancement by organic complexing agents under various conditions.

Keywords: Radionuclide, Solubility, Organic complexing agent, Support vector machine

Spectroscopic Properties of U(VI)-Acetate Species Near Neutral pH Conditions

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The bioreduction process from soluble U(VI) to insoluble U(IV) has been extensively studied in the field of radionuclides migration. Since acetic acid (AcOH) is widely used as an electron donor for bioreduction of U(VI), it is necessary to understand the effect of U(VI)-AcOH complexes that exist in different species depending on pH on this process. Changes in samples before and after bioreduction can be compared using time-resolved laser luminescence spectroscopy (TRLSS), which measures the characteristic luminescence spectra of different U(VI) species. Although luminescence properties of U(VI)-AcOH species were reported, experiments were conducted under conditions below pH 4.5. In this study, spectrophotometry and TRLSS for U(VI)-AcOH species (10–100 μ M U(VI) and 20 mM AcOH) were performed in pH ranges extending to neutral and alkaline pH regions similar to groundwater conditions as well as acidic pH region.

Two different complexes ($\text{UO}_2(\text{AcO})^+$, $\text{UO}_2(\text{AcO})_2$ with U(VI) and acetate ratios of 1:1, 1:2) were observed in the acidic pH region. The 1:1 complex, which appears as the pH increases, has no luminescence properties, but its presence can be confirmed because it serves to reduce the luminescence intensity of UO_2^{2+} . In contrast, the 1:2 complex exhibits distinct luminescence properties that distinguish it from UO_2^{2+} . The 1:3 complex ($\text{UO}_2(\text{AcO})_3^-$) expected to appear with increasing pH was not observed. Two different complexes ($(\text{UO}_2)_3(\text{OH})_5^+$, $(\text{UO}_2)_3(\text{OH})_7^-$ with U(VI) and OH ratios of 3:5, 3:7) were observed as the major species in the neutral pH region, but their luminescence lifetimes are remarkably short compared those in the absence of AcOH. Solid U(VI) particles were observed in the alkaline pH region, and they also had completely different luminescence properties from the aforementioned U(VI)-AcOH and U(VI)-hydrolysis species. Based on these results, the effect of pH in the presence of AcOH on the bioreduction process from U(VI) to U(IV) will be discussed.

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Keywords: Time-resolved laser luminescence spectroscopy (TRLSS), Bioreduction of U(VI), U(VI)-acetate complexes, U(VI)-hydrolysis complexes, Colloidal U(VI) particles

Radial Distribution of Noble Metal Precipitates in a Used Fuel Pin

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Noble metal precipitates are fission products that can be found as metallic alloys in used nuclear fuel. They do not exist homogeneously inside the fuel pellets, but exist in grain boundaries in the form of immiscible particles. The first drawback that comes because they exist in grain boundaries is the degradation of mechanical integrity. The particles in the grain boundaries can be considered as defects in a solid solution of uranium oxide pellets, and they can change the lattice volume. Therefore, it is known that it can cause stress corrosion cracking of fuel pellets. Furthermore, there is a negative effect from the perspective of used fuel management. However, they also have a positive effect on used fuel management. Since the noble metal has a galvanic reduction effect, the particles serve as an oxidation inhibitor for uranium.

There are many other effects regarding the noble metal precipitates. However, in any case, quantifying the particles is important in order to quantitatively analyze these effects from the perspective of used fuel management. SCALE/TRITON code was applied to calculate the noble metal isotopes including Mo, Tc, Ru, Rh and Pd. In order to calculate the distribution inside the pin, the multiregion cell model was selected. In particular, a cylindrical geometry was used, and the pellet was divided into several layers. In addition, coolant and cladding surrounded the pellet. Finally, the radial distribution was evaluated using the computational code, along with a neutron flux map.

Keywords: Used fuel, Noble metal precipitates, High-level waste, Grain boundaries

Analysis of ^{129}I Iodine in Cement Waste Form Including ^{137}Cs , ^{60}Co

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Korea Atomic Energy Research Institute (KAERI) is planning to disposal of the radioactive contaminated cement waste form to the final disposal facility. The final disposal facility require evaluation of immersion, compressive strength, and radionuclide inventory of radioactive wastes to meet the acceptance criteria for safe disposal. According to the LILW acceptance criteria of the Nuclear Safety and Security Commission of Korea (NSSC), the disposal limit radioactivity of ^{129}I ($3.70 \times 10^1 \text{ Bq/g}$) is lower than other radionuclides. ^{129}I emits low energy as its disposal limit is low, so it is difficult to analyze in the presence of ^{137}Cs and ^{60}Co which emit high energy. Therefore, it is essential to accurately separate and analyze iodine in radioactive waste.

In this study, we focused on the determination of ^{129}I in cement waste form containing ^{137}Cs , ^{60}Co . We added 1 g of ^{129}I (11.084 Bq), ^{137}Cs (1,300 Bq) and ^{60}Co (402 Bq) to cement waste form, respectively. The separation of ^{129}I in cement waste form was carried out using an acid leaching method. And, we confirmed the specific activity of ^{137}Cs and ^{60}Co at each separation step. It was observed that an acid leaching method showed the remove efficiency ^{137}Cs (99.97%) and ^{60}Co (99.94%), respectively. In addition, ^{129}I was also analyzed at approximately 96.44% in simulated contaminated cement waste form.

In conclusion, through this experiment, it was confirmed that ^{129}I could be successfully separated and analyzed by using the acid leaching method in cement waste form containing excessive ^{137}Cs and ^{60}Co .

Keywords: Iodine, Cement waste form, Acid leaching

Spectroscopic Study of Ternary Am(III)-ISA-OH Species and Their Effects on the Solubility of Am Under Alkaline Conditions

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Mobility of radionuclides (RNs) in natural water systems can be increased by complex formation with organic materials. In alkaline cement pore-water conditions, cellulose materials in radwastes such as woods and papers are degraded fast to small organic materials. As a major cellulose degradation product, isosaccharinate (ISA) has been paid attention recently due to its effect on facilitating RNs migration. ISA contains a carboxyl and four hydroxyl functional groups, which cooperatively interact to form chelating bonds with positively charged radionuclides. In our previous study, we determined thermodynamic formation constants, reaction enthalpy and entropy of trivalent americium complexes with ISA, $\text{Am}(\text{ISA})_n^{(3-n)+}$ ($n=1, 2$), in weak acidic condition by conducting temperature-dependent UV-Vis absorption spectroscopy. Based on those thermodynamic constants along with the experimental results from time-resolved laser induced fluorescence spectroscopy and DFT calculations, we suggested two different chelating-modes of ISA on Am(III). It is more relevant to study Am(III)-ISA complexation under alkaline conditions around pH 12.5, which correspond to the pore-water condition of calcium-silicate-hydrate. Under the alkaline conditions, deprotonated hydroxyl groups of ISA can form more strong interactions with Am. Aquatic hydroxide group can also act as a ligand to form ternary Am(III)-ISA-OH complexes. In this study, absorption spectra of Am-ISA systems were monitored with two variations: first, pH variation (5.5–13) in the presence of constant 30 mM ISA, and second, ISA concentration variation (20 μM – 30 mM) at constant pH of 12.5. As increasing the pH at constant 30 mM ISA, absorption spectra of $\text{Am}(\text{ISA})_2^+$ were red-shifted from 506.3 to 509.5 nm. The samples showed stable absorption spectra over 30 days. On the other hand, samples with lower ISA concentrations below 10 mM at pH 12.5, showed gradual decrease in the absorbance as sample aging time. By examining filtrates after ultrafiltration (1 kDa), we confirmed that aqueous Am(III)-ISA complexes were formed in the presence of 30 mM ISA at pH 12.5, while colloidal particles and precipitations were formed in the conditions of ISA concentrations lower than 10 mM. In this presentation, we will discuss about probable ternary complex forms of Am(III)-ISA-OH, colloidal forms, and solubility of Am(III) as a function of ISA concentration under alkaline conditions. Absorption and luminescence spectroscopic properties of the Am(III)-ISA-OH ternary system will also be presented.

Keywords: Americium, Isosaccharinate, Ternary complex, Colloidal stability, Solubility, Spectroscopy, Ultrafiltration

Evaluation of Solubility and Species Distributions of Americium in Synthetic KURT Groundwater by Geochemical Modeling and Experiments Under Oversaturation Conditions

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Solubility and species distributions of radionuclides in domestic groundwater conditions are required for the safety assessment of deep underground disposal system of spent nuclear fuel (SNF). Minor actinides including Am contribute significant extents to the long-term radiotoxicity of SNF. In this study, the solubility of Am was evaluated in synthetic groundwater (Syn-DB3), which were simulated for the groundwater of the DB3 site in the KAERI Underground Research Tunnel (KURT). Geochemical modeling was performed based on the ThermoChimie_11a (2022) thermochemical database from Andra to estimate the solubility and species distributions of Am in the Syn-DB3 condition. Dissolved Am concentrations in the Syn-DB3 were experimentally measured under oversaturation conditions. Am(III) stock solution in perchlorate media was sequentially diluted in Syn-DB3 to prepare 8 μM Am(III) in Syn-DB3. The pH of the solutions was adjusted to be in the range of 6.4–10.5. A portion of the samples was transferred to quartz cells for UV-Vis absorption and time-resolved laser fluorescence spectroscopy studies and the rest were stored in centrifuge tubes. The absorption spectra of the samples were monitored over 70 days and the results suggest that Am colloidal particles were formed initially in all the samples and precipitated rapidly within two days. Over the experimental period of 236 days, small volume (10 μL) of the samples in the centrifuge tubes were periodically withdrawn after centrifugation (18000 rpm, 1 hr) for the liquid scintillation counting to measure the concentrations of Am dissolved in Syn-DB3. In the end of the experiments, pH of the samples was checked again and the final dissolved Am concentrations were determined after ultrafiltration (10 kDa) to exclude the contribution of colloidal particles. In the pH range of 8-9, which is relevant to the KURT-DB3 groundwater condition, the measured dissolved Am(III) concentrations were converged to around 10^{-8} M. These values are higher than the solubility of $\text{AmCO}_3\text{OH}\cdot 0.5\text{H}_2\text{O}(\text{s})$, but lower than that of $\text{AmCO}_3\text{OH}(\text{am})$. There was no indication of transformation of the amorphous phase to the crystalline phase in our observation time window.

Keywords: Americium, KURT groundwater, Solubility, Geochemical modeling, Thermochemical data, Ultrafiltration

Factors Affecting the Effectiveness of Zinc Injection in Nuclear Power Plants

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Zinc injection into the coolant system of nuclear power plants is an effective method for reducing corrosion and improving performance. The effectiveness of this method is influenced by various factors such as zinc concentration and injection rate.

This paper provides an overview of the factors affecting the effectiveness of zinc injection in nuclear power plants, with a focus on zinc concentration and injection rate, and discusses various research results on the effects of these factors on corrosion reduction and coolant system performance.

Zinc concentration is an important factor affecting the effectiveness of zinc injection. The research results show that gradual increases in zinc concentration are more effective for coolant system stability. However, the concentration should not exceed the recommended levels as high zinc concentrations can have negative effects on the system.

Injection rate is also an important factor affecting the effectiveness of this method. The research results show that gradual increases in injection rate are more effective for coolant system stability. However, excessive injection rates can have negative effects on the system such as overload of the zinc injection facility and chemical shocks within the coolant system, and therefore, should be optimized.

In conclusion, zinc concentration and injection rate are important factors affecting the effectiveness of zinc injection in nuclear power plants. The optimal concentration and injection rate should be determined based on specific reactor conditions and system requirements, and efforts should be made to maximize corrosion reduction and performance improvement.

Keywords: Zinc injection, Injection rate, Zinc concentration, Nuclear power plant

Prediction of the Mobility of ^{99}Tc , ^{137}Cs , and ^{238}U Through the Engineered and Natural Barriers Using Support Vector Machine

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A low- and intermediate-level radioactive waste repository contains different types of radionuclides and organic complexing agents. Their chemical interaction in the repository can result in the formation of radionuclide-ligand complexes, leading to their high transport behaviors in the engineered and natural rock barriers. Furthermore, the release of radionuclides from the repository can pose a significant risk to both human health and the environment. This study explores the impact of different experimental conditions on the transport behaviors of ^{99}Tc , ^{137}Cs , and ^{238}U through three types of barrier samples: concrete, sedimentary rock, and granite. To assess the transport behavior of the samples, the geochemical characteristics were determined using X-ray diffraction (XRD), X-ray fluorescence (XRF), Fourier-transform infrared spectroscopy (FTIR), scanning electron microscopy with energy-dispersive X-ray spectroscopy (SEM-EDS), and Brunauer-Emmett-Teller (BET) analysis. The adsorption distribution coefficient (K_d) was used as an indicator of transport behavior, and it was determined in batch systems under different conditions such as solution pH (ranging from 7 to 13), temperature (ranging from 10 to 40°C), and with the presence of organic complexing agents such as ethylenediaminetetraacetic acid (EDTA), nitrilotriacetic acid (NTA), and isosaccharinic acid (ISA). A support vector machine (SVM) was used to develop a prediction model for the K_d values. It was found that, regardless of the experimental parameters, ^{99}Tc may migrate easily due to its anionic property. Conversely, ^{137}Cs showed low transport behaviors under all tested conditions. The transport behaviors of ^{238}U were impacted by the order of EDTA > NTA > ISA, particularly with the concrete sample. The SVM models can also be used to predict the K_d values of the radionuclides in the event of an accidental release from the repository.

Keywords: Radionuclides, Sorption distribution coefficient, Organic complexing agent, Support vector machine

Anodic Dissolution Behavior of Zr-Nb Alloy in Chloride-based Molten Salts

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Zirconium(Zr) alloys are commonly used in the nuclear industry for applications such as fuel cladding and pressure tubes. To minimize the levels and volumes of radioactive waste, molten salts have been employed for decontaminating Zr alloys. Recently, a two-step Zr metal recovery process, combining electrolysis and thermal decomposition, has been proposed.

In the electrolysis process, potentiostatic electrorefining is utilized to control the chemical form of electrodeposits(ZrCl). Although Zr metals are expected to dissolve into molten salts, reductive alloy elements can also be co-dissolved and deposited on the cathode. Therefore, a better understanding of the anodic side's response during potentiostatic electrorefining is necessary to ensure the purity of recovered Zr and long-term process operation.

As the first step, potentiodynamic polarization curves were obtained using Zr, Nb, and Zr-Nb alloy to investigate the anodic dissolution behavior in the molten salts. Nb, which has a redox potential close to Zr, and Zr exhibit active or passivation dissolution mechanisms depending on the potential range. It was confirmed that Zr-Nb alloy also has a passivation region between -0.223 to -0.092 V influenced by the major elements Zr and Nb. Secondly, active dissolution of Zr-Nb was performed in the range of -0.9 to -0.6 V. The dissolution mechanism can be explained by percolation theory, which is consistent with the observed microstructure of the alloy.

Thirdly, passivation dissolution of Zr, Nb, and Zr-Nb alloy was investigated to identify the pure passivation products and additional products in the Zr-Nb alloy case. K_2ZrCl_6 and K_3NbCl_6 were identified as the pure passivation products of the major elements. In the Zr-Nb alloy case, additional products, such as Nb and NbZr, produced by the redox reaction of nanoparticles in the high viscous salt layer near the anode, were also confirmed.

The anodic dissolution mechanism of Zr-Nb alloy can be summarized as follows. During active dissolution, only Zr metal dissolves into molten salts by percolation. Above the solubility near the anode, passivation products begin to form. The anode potential increases due to the disturbance of passivation products on ion flow, leading to co-dissolution of Nb. When the concentration of Nb ion exceeds the solubility, a passivation product of Nb also forms. In this scenario, a high viscous salt layer is formed, which traps nanoparticles of Zr metal, resulting in redox behavior between Zr metal and Nb ion. Some nanoparticles of Zr and Nb metal are also present in the form of NbZr.

Keywords: Zr-Nb alloy, Electrorefining, Anodic dissolution, Passivation, Molten salts

Establishment Status of Radionuclide Analysis System for Below the Low-level Decommissioning Radioactive Wastes at KRID

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Support for nuclear power plant (NPP) dismantling & decommissioning (D&D) industry is necessary through development of the infrastructure and the D&D technology. Because KORI#1 and Wolsong#1 is planned to decommission until around 2030.

Korea research institute of decommissioning (KRID) was established through the preliminary feasibility study. KRID has plan to support nuclear companies to join D&D industry. Normal facilities (Lv.1) of KRID infracstucture are currently being constructed and radiation management facilities (Lv.2) construction is expected to begin in October. Further, KRID is planning the construction of equipment to develop the procedure for radionuclide analysis through R&D project. A total period of the R&D project is 45 months, and the total R&D funding for this period is 19.4 billion won. The ultimate goal of the R&D project is to build the infractstucture base to analyze decommissioning radioactive wastes. Furthermore, the R&D project is important to reliably perform the NPP D&D.

Keywords: Decommissioning, Radionuclide analysis, R&D project, KRID

The Dehydration Process and Evaluation of NaCl-MgCl₂ Using Elemental Mg

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Molten salt reactors have several advantages over conventional light water reactors. These include producing less nuclear waste, operating at higher power efficiency and inherent safety due to the low operating pressure. NaCl-MgCl₂ eutectic salt is one of the candidates for the molten salt reactor coolant. However, because the salt is very hygroscopic, structural material corrosion occurs resulting in the high cost to maintain. To mitigate corrosion there have been many studies for the dehydration of the salt, especially focusing on the magnesium chloride. The reason is that the moisture adsorbed to the magnesium chloride undergoes hydrolysis over 200 degrees Celsius and decomposes to MgOHCl while the moisture associated with the NaCl is easily liberated during the heating procedure without chemical reaction. As the operating temperature of the molten salt is between 500 and 700 degrees Celsius, the MgOHCl is believed as the main cause for the structural corrosion. In this research, thermal dehydration of the salt with elemental Mg, for the NaCl-MgCl₂ eutectic, was studied based on the previous dehydration methods and considering scalable and easy to handle. The MgOHCl was removed both through the thermal decomposition and the reduction by Mg metal. After the removal of MgOHCl, based on the difference between the freezing points and the density, the salt cooled down very slowly to ensure the separation between the purified salt and the disposals such as MgO and remaining Mg metals. The efficiency of the dehydration method was determined by the concentration of the MgOHCl. The concentration was determined by cyclic voltammetry and the result was compared with undehydrated salt and salt dehydrated thermally without the addition of Mg metal. To qualify and quantify the MgOHCl content through the cyclic voltammetry, it was necessary to observe the signal by adding MgOHCl to each sample. Based on the thermogravimetric analysis result of MgCl₂·6H₂O, MgOHCl powder was formed through heating the MgCl₂·6H₂O.

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Keywords: Thermal, Hydrolysis, MgOHCl, Cyclic voltammetry, Thermogravimetric analysis

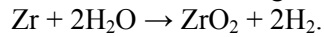
Change in Composition of Zirconium Under the Pressurized Water at High Temperature

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Change in chemical constitution of a zirconium sample treated at the simulated condition of a pressurized water reactor (water at 315°C and 15.5 MPa) was investigated using X-ray diffraction (XRD) and Raman spectroscopy. We observed swelling of the zirconium sample as well as the change in its color from silver to gray treated after the pressurized water at high temperature. On the basis of XRD and Raman data, we confirmed that the variation in composition of zirconium specimen from hexagonal Zr to monoclinic ZrO₂ occurred at the simulated PWR condition. Therefore, we suggested that the oxidation of zirconium appeared due to the reaction with water at high temperature and pressure, as shown in the following reaction:



Keywords: Zirconium, Cladding, Oxidation, Pressurized water reactor, High temperature and pressure

Thermal Behavior of Epsilon Particles in Uranium Oxide

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Confirmation of the thermal behavior of spent fuel is one of the important points in the management of high-level radioactive waste. This is because various fission products exist in spent nuclear fuel, and a management plan according to their behavior is required. Among the fission products, epsilon particles exist in the form of metal deposits and have a great influence on their physical and chemical properties. However, observing the thermal behavior of epsilon particles is important for understanding spent fuel behavior in thermally environment, but it is difficult to maintain a consistent thermal environment.

In this work, we report the thermal behaviors study of uranium oxide with epsilon particle using in situ high temperature X-ray diffraction. We measured the variation of temperature on the size of crystalline, which is a cell parameter in the reaction process. And then, the change of lattice parameters is calculated by Rietveld refinement.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (2021M2E3A3040092).

Keywords: Thermal behavior, Epsilon particle, Uranium oxide, High-temperature X-ray diffraction

Viscosity of Molten Salt Systems: NaCl-MgCl₂ and NaCl-MgCl₂-UCl₃

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Viscosity of molten salts is an essential property for the thermal hydraulic design and evaluation of molten salt reactor (MSR). Therefore, viscosity data is one of the fundamental physical property data required for safe process operation and countermeasures to severe accidents. In this study, based on our experience of developing a viscosity measurement system for high-temperature LiCl-KCl molten salt system, the viscosity of NaCl-MgCl₂ and NaCl-MgCl₂-UCl₃ molten salts, which are considered promising salts in MSR, was measured. In order to investigate the physical properties of uranium in high-temperature NaCl-MgCl₂ molten salt, a viscometer system for high-temperature viscosity measurement was specially designed. As a result of the measurement, the viscosity of the 58NaCl-42MgCl₂ molten salt was 2.73 cP at 838 K, 2.15 cP at 889 K, and 1.68 cP at 940 K. And the viscosity of 73NaCl-21MgCl₂-6UCl₃ molten salt was 3.79 cP at 877 K, 3.58 cP at 897 K, and 1.63 cP at 941 K. The repeatability of the measurement showed a precision of less than 3%. Although sufficiently-verified starting materials were not used, viscosity data were reported for the first time for NaCl-MgCl₂-UCl₃ molten salts.

Keywords: Molten salt, Viscosity, Sodium chloride, Magnesium chloride, Uranium chloride

Corrosion Rate of Ni-based Alloys in LiCl Molten Salt

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In molten salt reactor (MSR), liquid fuel integrated with the molten salt coolant is used to improve the safety, resulting in the prevention of the loss-of-coolant accident (LOCA) that can occur in a pressurized water reactor (PWR). Because the structural materials used in MSRs are directly contacted with liquid fuel for a long time, they must have excellent corrosion resistance to the molten salt. Therefore, to examine the corrosion rates for Ni-based alloys in the molten salt, the corrosion experiments for alloy 600, alloy 617, and Hastelloy N were performed under LiCl molten salt at 635°C for 100 h in a glove box under Ar environment. Through a weight loss method for the three Ni-based coupons before and after the corrosion tests, we evaluated their corrosion rates. Based on the results of weight loss for each alloy, we confirmed that Hastelloy N has the excellent corrosion resistance compared to the other alloys. Furthermore, the changes in the crystal structure and surface morphology with elemental distribution for the three alloys by corrosion in LiCl molten salt were analyzed, showing the variation in surface topography and the decrease in Cr element after corrosion experiments for all coupons.

Keywords: Ni-based alloys, Molten salt reactor, Hastelloy N, Corrosion rate, LiCl molten salt

Molten Salt Densities of NaCl-MgCl₂-UCl₃ System

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The density of molten salts is the most important property in the development of molten salt reactor (MSR). The density value measured through the experiment is also very valuable as a gold standard for the validation of the prediction models based on molecular dynamics or other computational methods. To the best of our knowledge, the experimental density data of the ternary NaCl-MgCl₂-UCl₃ salt system as a MSR candidate fuel salt have never been reported previously. In this study, density measurement experiment of high-temperature molten salt of NaCl-MgCl₂ and NaCl-MgCl₂-UCl₃ was conducted using a previously-developed density measurement system based on the maximum bubble pressure (MPB) method. As a result of the experiment, the density value of 62NaCl-18MgCl₂-20UCl₃ molten salt at 873 K was 2.62 g/cm³. A density prediction value of 2.65 g/cm³ at 873 K was derived from the obtained results based on the rule of additivity of molar volume method. The predicted density of 62NaCl-18MgCl₂-20UCl₃ was consistent with the experimental value within 1%. The density measuring system used in this study is promising for the validation of other multi-component molten salt systems.

Keywords: Molten salt, Density, Sodium chloride, Magnesium chloride, Uranium chloride

Effects of Heat Treatment on Corrosion Behaviors of 3-D Printed Ni-based Alloy in Molten NaCl-MgCl₂ Salt

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Molten salt is one of the promising medium materials for molten salt reactors and energy storage systems. Molten salt is advantageous for better physical properties such as low melting point and high boiling point, high energy capacity, high thermal conductivity, and high thermal stability than other medium materials such as water or liquid metals. However, the corrosivity of the molten salt is one of the main factors that disturbs the various applications of the molten salt.

On the other hand, metallic 3-D printing technologies have developed by leaps and bounds over the past 20 years and show potential for use in cutting-edge industries such as aerospace and military purposes. However, the biggest problem of 3-D printed products is that the mechanical and physical properties are very weak along the laminated plane that was generated during the manufacturing process. In particular, other research showed that corrosion is vulnerable through the laminated surface, and corrosion along the laminated plane is not completely mitigated through a general heat treatment process although the microstructure of the surface is evaluated to be partially mitigated by the heat treatment.

In this study, molten salt corrosion behaviors of simple Ni-based alloy with a composition of 80Ni-20Cr were analyzed. Ni-based alloys were fabricated by casting and 3-D printing, and some of the 3-D printed specimens were thermally treated at 1,273 K for 1 hour to examine the effects of heat treatment on corrosion behaviors. In molten eutectic NaCl-MgCl₂ melts at 973 K, Ni-based alloys were corroded for 1, 3, 7, and 28 days and their microstructural changes were analyzed by SEM-EBSD-EDS and OM. The corrosion behaviors of the alloy were also evaluated by the salt composition measured with ICP-OES. 3-D printed alloy with post-treatment showed more resistivity to the molten salt corrosion than as-fabricated 3-D printed alloy. However, the corrosion rate of the 3-D printed specimen after heat treatment was still higher than that made by casting.

Acknowledgements

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Keywords: Molten salt corrosion, Ni-based alloy, 3-D printing, Heat treatment, NaCl-MgCl₂

Study of Pu(III) and Isosaccharinate Complexation Using UV-Vis Absorption Spectrophotometry

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Plutonium exhibits a variety of oxidation states and has a strong affinity for complexation with organic ligands. Isosaccharinic acid (ISA) is a major degradation product of cellulose materials present in the low to intermediate radioactive wastes. The interaction between trivalent plutonium and ISA can significantly impact the migration and containment of plutonium in the repository environment. In this study, formation of Pu(III) and ISA complexes was investigated at an ionic strength of 1 M of NaClO₄ using UV-Vis absorption spectrophotometry. To exclude the effect of the Pu(III) oxidation, absorption spectra were measured within 10 min after adding ISA into Pu(III) solution and processed using HYPSPPEC software for deconvolution after baseline correction. Several previous studies showed that the presence of ligands accelerates the oxidation of Pu(III) to Pu(IV). To investigate whether ISA complexation can also accelerate the Pu(III) oxidation, UV-Vis absorption spectra changes over 24 hours were analyzed as a function of the ratio of ISA to plutonium concentration.

Keywords: Trivalent plutonium, Isosaccharinic acid, Complexation, Spectrophotometry, Plutonium oxidation

Measuring the Performance of a Microelectrode Over Time in LiCl-KCl Eutectic at Different Temperatures

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Measuring the concentration of corrosion products or nuclear fission products (FPs) in molten salts is crucial for pyroprocessing and molten salt reactors. Electrochemical analysis methods that can be performed in situ offer significant advantages for monitoring the concentration of corrosion products or FPs in molten salts. A microelectrode is an electrode with a length of several tens of micrometers on one side. The use of a microelectrode for electrochemical analysis has several advantages due to its small size, including rapidly reaching the limiting current regardless of the scan rate, immediate attainment of the limiting current upon applying an overpotential for instant monitoring within milliseconds, accurate measurement even in low convection systems, a small iR drop resulting from low flowing current and high signal accuracy, and high current density resulting in a high signal-to-noise ratio (SNR). Among various methods for making microelectrodes, techniques involving cutting a thin wire or using capillaries (such as the dual-bore capillary and pulled glass capillary methods) require precise manual skills and experience. Therefore, the results may vary depending on the maker's skill level, and it can be difficult to control the electrode's area, thickness, and surface uniformly. Recent research has focused on using semiconductor processes to fabricate microelectrodes, where CVD, metal sputtering, photolithography, and etching processes work together to deposit, refine, and shape the required material on a silicon wafer to create microelectrodes. However, the durability of microelectrodes produced this way is still low (usable for about 15-30 minutes), and there is no clear research on the degradation mechanism over time. To verify the proper operation of the fabricated microelectrodes, cyclic voltammetry (CV) is performed at various scan rates (from 10 mVs⁻¹ to 2 Vs⁻¹), and chronoamperometry (CA) is also examined to confirm whether the electrodes rapidly reach a steady-state current. After confirming their proper operation, CV is continuously measured until the microelectrodes are destroyed in a LiCl-KCl solution containing a small amount of FPs (Sm 340 mM) at 450°C. By observing changes in the electrical signal of the microelectrodes over time, the durability is evaluated, and the mechanism of performance degradation of the electrode is discovered. The experiment is then repeated by gradually increasing the temperature by 30°C from 450°C up to 600°C to observe the changes with temperature. This study provides basic information for future microelectrode experiments, and by diagnosing the cause of destruction, a more durable microelectrode structure can be manufactured.

Acknowledgements

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Keywords: Pyroprocessing, Electrochemical monitoring, Cyclic voltammogram, Chronoamperometry, Microelectrode, Semiconductor process

A Study on the Effect of Sample Matrix on Gamma Spectrometry of Radioactive Sludge Samples

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To analyze the activity concentration of radionuclides in radioactive sludge samples generated from low- and intermediate-low-level radioactive waste from domestic nuclear power plant, a pretreatment process that dissolves and homogenizes the sample is essential. However, this pretreatment process requires the use of hydrofluoric acid, which makes analysis difficult and challenges users to handle harmful chemicals. Therefore, we aim to minimize the use of hydrofluoric acid by measuring gamma nuclides in the sludge sample without pretreatment process and compare the differences of measurement results according to the sample matrix with and without pretreatment process. We will collect about 0.1 g of the sludge sample, and dissolve it using an acid treatment process after using microwave decomposition. We will then use gamma spectroscopy to check the concentration of nuclides present in the sludge before and after dissolution and consider the effect of the sample matrix.

Keywords: Radioactive waste, Gamma spectrometry, Radiochemical analysis, Radioactive sludge

A Study on the Evaluation of Summing Coincidence Effect for Gamma Nuclide Analysis in Radioactive Waste

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Many radionuclides emit two or more gamma rays in a cascade once they decay. At this time, gamma rays are detected at the same time, and the signals are overlapped and measured as one added signal. This is called the summing coincidence effect, and it causes an error of more than 10% depending on the detection efficiency, measurement conditions, and target nuclide. It is known to be greater as the efficiency of the detector increases and as the distance between the source and the detector decreases. It is necessary to consider the summing coincidence effect since the efficiency of the HPGe detector owned by the KHNP CRI is as high as 65%. In this study, We would like to propose an appropriate gamma nuclide analysis method for radioactive waste generated from NPP by evaluating the influence on the summing coincidence effect.

Keywords: Summing coincidence effect, TCS, HPGe, Cascade

Gross-beta Measurement Method Comparison of Liquid Scintillation Counter and Proportional Gas Counter

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When self-disposing of radioactive waste, it is important to follow the acceptable concentration standards for each nuclide set by the Nuclear Safety and Security Commission (NSSC). Gamma-emitting nuclides can be easily analyzed with a simple pretreatment process, but beta-emitting nuclides require a chemical separation procedure to be analyzed for radiochemistry analysis. When analyzing beta-emitting nuclides for the purpose of self-disposal, there may be difficulties in radiation detection after the chemical separation process. This is because the concentration of beta nuclides in the sample may be low and some of them may be lost during the chemical separation. Therefore, measurement method of gross-beta activity can be used instead of that of each nuclide to access the compliance of self-disposal criteria. While a proportional counter is commonly used to measure gross-beta activity, liquid scintillation counting can also be used to measure gross-beta, and we plan to compare the results of both methods.

Keywords: Radioactive waste, Self-disposal, Gross-beta, LSC

A Study on the Effects of Background Increase by Compton Continuum of Co-60

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For the final disposal of radioactive waste, concentration of gamma nuclides such as Co-58, Co-60, Cs-137, Nb-94 have to be determined to meet nuclear regulatory requirements. In general, gamma nuclide analysis can be performed with simple sample pretreatment without complicated chemical separation processes due to the characteristics of the nuclide and high resolution of the measuring equipment.

However, when the concentration of Co-60 is high in a specific radioactive waste generated at the NPP, the background is increased by the Compton continuum of Co-60. That makes it difficult to evaluate accurately Nb-94, which is in the lower energy band than the gamma ray energy region of Co-60 and especially Cs-137, which is used as a key nuclide of scaling factor.

In this study, We consider the problem of MDA dissatisfaction or overestimation due to the increased background by Co-60.

Keywords: Compton continuum, Radioactive waste, Gamma nuclide

Separation of Light Rare Earth Elements and High Activity Nuclides Using Gas Pressurized Extraction Chromatography With Anion Exchange Resin

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This study was performed to evaluate the separation of Sr, Cs, Ba, La, Ce, and Nd using gas pressurized extraction chromatography (GPEC) with anion exchange resin for the quantitation of Neodymium. GPEC is a micro-scaled column chromatography system that provides a constant flow rate by utilizing nitrogen gas. It is overcome the disadvantages of conventional column chromatography by reducing the volume of elution solvent and shortening the analysis time. Here, we compared the conventional column chromatography and the GPEC method. The whole analysis time was decreased by nine times and radioactive wastes were reduced by five times using the GPEC system.

Anion exchange resin 1-X4 (200~400 mesh size) was used. The sample was prepared at a 0.8 M nitric acid in methanol solution. The elution solvent was used at a 0.01 M nitric acid in methanol solution. Finally the eluate was analyzed by ICP-MS to determine the identification and recovery. In this case, we applied the natural isotopes of LREEs (¹³⁹La, ¹⁴⁰Ce, and ¹⁴⁴Nd) and high activity nuclides (⁸⁸Sr, ¹³³Cs, and ¹³⁸Ba) instead of radioactive isotopes for the preliminary test; as a result, unnecessary radioactive waste was not produced. The recoveries were 93.9%, 105.9%, 91.9%, 47.6%, 35.9%, and 79.9% of Sr, Cs, Ba, La, Ce, and Nd, respectively. The reproducibility of recoveries by GPEC were in the range 2.8%–10.9%.

Keywords: Gas pressurized extraction chromatography, Column chromatography, Anion exchange resin, Light rare earth elements, High activity nuclides

A Study on Thermal Decomposition Method for Simultaneous Analysis of ^3H and ^{14}C in DAW Radioactive Wastes

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To analyze the radioactivity of ^3H and ^{14}C in miscellaneous radioactive wastes generated from nuclear power plants, a wet digestion method using sulfuric acid is currently used. However, sulfuric acid is classified as a special management material, and there is no disposal method for contaminated radioactive waste. Therefore, research on a thermal decomposition method that can analyze the DAW radioactive waste samples without using sulfuric acid is necessary. In this study, we will cover the final sample amount, sample injection method, and prevention of organic ignition to meet the minimum detection limit requirements of the analysis equipment. Through this research, optimal conditions for the thermal decomposition method for analyzing the radioactivity of ^3H and ^{14}C in DAW radioactive wastes generated from nuclear power plants can be derived.

Keywords: DAW, Activity, ^3H , ^{14}C

Evaluating Dead Time and Dilution Factors for Improving Radioactive Waste Analysis

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The Korea Atomic Energy Research Institute (KAERI) employs a methodology for evaluating the concentration of radionuclides, dividing them into volatile and non-volatile nuclides based on their characteristics, to ensure the permanent disposal of internally generated radioactive waste. Gamma spectroscopy enables the detection and radiation concentration determination of individual nuclides in samples containing multiple gamma-emitting nuclides. Due to the stochastic nature of radioactive decay, the generated radiation signal can interact with the detector faster than the detected signal processing time, causing dead time in the gamma spectroscopy process.

Radioactive waste samples typically exhibit higher radiation levels than environmental samples, leading to long dead times during the measurement process, consequently reducing the accuracy of the analysis. Therefore, dead time must be considered when analyzing radioactive waste samples. During the measurement process, dead time may vary between a few seconds to several tens of thousands of seconds. More long dead time may also result in a temporal loss in the analysis stage, requiring more time than the actual measurement time. Long dead time samples undergo re-measurement after dilution to facilitate the analysis. As the prepared solution is also utilized in the nuclide separation processes, minimizing sample loss during dilution is crucial. Hence, predicting the possibility of dead time exceeding the target sample in advance and determining the corresponding dilution factor can prevent delays in the analysis process and the loss of samples due to dilution.

In this study, to improve the issues related to gamma analysis, by using data generated during the analysis process, investigated methods to predict long dead time samples in advance and determining criteria for dilution factors.

As a result of comparing the dead time data of 5% or long with the dose of the solution sample, it was concluded that analysis should be performed after dilution when it is about 0.4 $\mu\text{Sv/h}$ or high. However, some samples required dilution even at doses below 0.4 $\mu\text{Sv/h}$. Also, re-measurement after dilution, the sample with a dead time of less than 32% was measured with less than 5% when diluted 10 times, and more than 32% required more than 10 times dilution. We suppose that with additional data collection for analyzing these samples in the future, if we can establish clearer criteria, we can predict long dead time samples in advance and solve the problem of analysis delay and sample loss.

Keywords: Radioactive waste, Gamma spectrometry, Radionuclide analysis, Dead time

Development of an Automatic Sampling Device for High-Temperature Molten Salt and Optimization of Sampling Conditions

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LiCl-KCl eutectic possesses unique properties such as a low melting point, high thermal conductivity, and good electrical conductivity. These properties make it suitable for various applications, including nuclear power generation, pyroprocessing in nuclear waste management, and thermal energy storage systems. In most experiments using LiCl-KCl, the molten salt composition is an important factor; therefore, periodic analysis through sampling is necessary for monitoring compositional changes. Although manual sampling is typically used, it is time-consuming and can introduce errors due to low reproducibility. To address this issue, we have developed an automatic molten salt sampling device using the cold-finger method. This method involves immersing the tip of a tungsten rod in high-temperature LiCl-KCl, removing it after a few seconds, and allowing the adhered molten salt to solidify instantly. A collector then scratches and drops the solidified sample. These processes are carried out automatically using servo motors, enabling the sampling device to move around the molten salt system. We have optimized the sampling conditions, such as insertion and withdrawal rate, immersion time, and the interval between continuous sampling, based on the molten salt temperature. The temperature was set between 500°C and 850°C, considering the operating temperatures of the applications. In addition to sampling speed, the sampling depth is a key condition for determining the sampling mass. Therefore, we examined the amount of sample depending on the sampling depth and, particularly, considered the change in salt height when sampling is performed continuously. As a result, we determined the number of sampling iterations required to reach the target sample mass. Furthermore, to minimize the initial salt loss, we noted that sampling from the salt surface resulted in less representative samples. To determine the reliability, we compared the results of surface sampling with those obtained when sampling at the middle of the salt. This study will enable highly reproducible and reliable sampling by providing a prototype for an automatic sampling device for molten salt along with guidelines.

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Keywords: Automatic sampling, Molten salt analysis, Cold-finger method

Uniformity of Droplet of Eutectic NaCl-MgCl₂ System for Corrosion Test of Coating Layer

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Corrosion-related challenges remain a significant research topic in developing next-generation Molten Salt Reactors (MSRs). To gain a deeper understanding of preventing corrosion in MSRs, previous studies have attempted to improve the corrosion resistance of structural alloys by coating surfaces such as alumina coating. To conduct a corrosion test of coating alloys fully immersed in molten salt, it's important to ensure that the coating application process is carefully carried out. Ideally, coating all sides of the alloy is necessary to avoid gaps like corners of the alloy, while only applying a one-sided coating alloy can lead to galvanic corrosion with the base metals. Using the droplet shape of eutectic salt applied to only one side of the coating alloy would avoid these problems in conventional corrosion immersion tests, as corrosion would occur solely on the coating surface. Although the droplet method for corrosion tests cannot fully replicate corrosion in the MSRs environment, it offers a valuable tool for comparing and evaluating the corrosion resistance of different coating surfaces of alloys. However, the surface area is important due to the effect of diffusion in the corrosion of alloy in molten salt environments, but it is difficult to unify in the case of droplet tests. Therefore, understanding the droplet-alloy properties and corrosion mechanism is needed to accurately predict and analyze these test systems' behavior highlighting unity for corrosion tests of different coating surfaces of alloys.

To analyze the molten salt droplet behavior on various samples, pelletized eutectic NaCl-MgCl₂ was prepared as salt and W-, Mo-coating, and base SS316 as samples. At room temperature, the same mass of pelletized eutectic NaCl-MgCl₂ was placed on different samples under an argon atmosphere and heated to a eutectic point of 500°C in a furnace. After every hour, the molten droplets were hardened by rapid cooling at room temperature outside the furnace. The mass loss of salts and the contact area of the samples were measured by mass balance and SEM. The shape, surface area to volume ratio, and evaporation of the droplets of NaCl-MgCl₂ per each coating sample and hour were analyzed to identify the optimal mass to equalize the contact coating surface of alloys with salts. Furthermore, We also analyzed whether their results reached saturation of corrosion products through ICP-MS. This will be significant research for the uniformity of the liquid-drop shape corrosion test of the coating sample in molten eutectic salts.

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Keywords: Coating, NaCl-MgCl₂ eutectic, Droplet corrosion

Extraction of Uranium Using Thermal Discharge of Domestic Nuclear Power Plants

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Given the limited terrestrial reserves of uranium (approximately 4.6 million tons), exploring alternative resources is necessary to secure a sustainable, long-term supply of nuclear energy. Uranium extraction from seawater (UES) is a potential solution since the amount of uranium dissolved in seawater (approximately 4.5 billion tons) is about 1,000 times that of terrestrial reserves. However, due to the ultra-low concentration of uranium in seawater (approximately 3.3 ppb), making UES economically viable is a challenging task.

In this paper, we explore the potential of using thermal discharge from domestic nuclear power plants for uranium extraction. The motivation for this comes from previous research showing that the adsorption capacity of amidoxime-based adsorbents is proportional to the temperature of the seawater in which they are deployed. Specifically, a study conducted in Japan found that a 10°C increase in seawater temperature resulted in a 1.5-fold increase in adsorption capacity.

Keywords: Uranium, Thermal discharge, Nuclear power plants, Extraction, Adsorbent

High-temperature X-ray Diffraction Study on Structure of Ni-based Alloys

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Crystallographic properties of Ni-based alloys such as alloys 600, 617, and Hastelloy N, which are a candidate to be used as structural materials in Molten Salt Reactor (MSR), were studied in the temperature range of 25-1,000°C using high-temperature X-ray diffraction (HT-XRD) under an Ar atmosphere. We found that face-centered cubic Ni crystal structure at room temperature was started to be changed over 600°C in all Ni-based samples. However, the appearance of changing diffraction patterns over 600°C was different for all samples. In addition, we observed the increase in the lattice constant along the a-axis upon heating in all specimens, determined by Pawley refinement of HT-XRD data.

Keywords: High-temperature X-ray diffraction, Molten salt reactor, Ni-based alloy, Lattice constant, Pawley refinement

Analysis of Tritium Compounds in Consumer Goods Using High Precision Gas Mass Spectrometry

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Some consumer goods containing radioactive substances are in circulation and used in everyday life. In accordance with the Nuclear Safety Act, consumer goods with radioactivity are regulated. However, since most consumer goods distributed in Korea have no information that can confirm the amount of radiation, it is necessary to analyze the radiation for safety regulation. Among these consumer goods, GTLS (Gaseous Tritium Light Source) contains gaseous tritium (tritium, written as ^3H or T), which is a radioactive material. The gaseous composition ratio in GTLS was analyzed using a precision gas mass spectrometer (Thermo Fisher, model MAT 271).

As a result of GTLS analysis, the H_2 , HD or $\text{H}_3^+(\text{T})$ or ^3He , HT or D_2 or He , DT , and T_2 , which correspond to the mass-to-charge ratio (m/z) 2 to 6 and the air components were detected. In addition, substances corresponding to $m/z=24$ and $m/z=21$ were also detected. These were compared with pure CH_4 and those fragmentation patterns. The ratios of CT_4 ($m/z = 24$) to CT_3 ($m/z = 21$) and CH_4 ($m/z = 16$) to CH_3 ($m/z = 15$) were compared and they agree within the measurement uncertainty. We also performed additional experiments to separate the water component in the GTLS samples, considering the possibility that the $m/z = 21$ to $m/z = 24$ region is tritium compounds based on H_2O . Despite the removal of the water components, peaks were detected at $m/z=21$ and $m/z=24$. Therefore, we confirmed that the component of $m/z = 24$ in the GTLS sample was CT_4 .

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Keywords: Tritium, Hydrogen isotopes, Gas measurement, Consumer goods

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