

2021 **추계** 학술대회 논문요약집

11. 17(수) ~ 19(금) 하이원그랜드호텔컨벤션타워



KOREAN
RADIOACTIVE WASTE
SOCIETY
AUTUMN 2021



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

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1분과

핵주기정책·규제 및 비확산(Oral)

Reverse Estimation for Particle Deposition of UO_2F_2 (holdup) Using Numerical Simulation: Survey and Research Status

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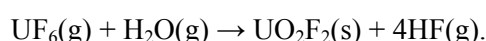
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Natural uranium is enriched as a form of the UF_6 gas. In this process, unexpected H_2O can penetrate into the gas pipe and react with UF_6 . This reaction produces UO_2F_2 in a solid phase based on the below reaction formula;



The UO_2F_2 produced here is the main material of holdup, which flows along the pipeline and precipitates onto the pipe wall. The physical properties (e.g., thickness or radioactivity) of the holdup can be a technical evidence of uranium enrichment. Holdup is an important phenomenon in terms of nuclear safeguards because it is a factor that can affect the transparency of nuclear material accountability.

There are various studies conducted to measure the thickness of holdup inside the pipe. However, studies to characterize holdup generated in various environments have been rarely conducted. In order to backup this technical gap, raw data under various operating conditions of an enrichment facility are required based on the assumption that the holdup generation is a function of the flow rate, the water vapor penetration rate, the pipe shape, and time. Then, the feasibility of reverse inference for the enrichment operating conditions needs to be analyzed.

To achieve this goal, we proposed new methodology which consists of two parts; The first part is to analyze the phenomenon on how holdup is adhered on the pipe wall by numerical simulation using COMSOL software. The second part is the radiological evaluation of the holdup effect generated inside the pipe. In the second part, the MCNP code is utilized. In this study, the survey and research status in this topic is described and preliminary results for the reverse estimation of the holdup using numerical simulation are presented.

Keywords: Holdup, Uranium enrichment, Nuclear safeguards, Nuclear material accountancy, Nuclear forensic, Nonproliferation

Analysis of White Papers Revision on PR&PP Characteristics of GEN-IV Nuclear Energy Systems

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The Generation IV International Forum (GIF) has been developing six types of advanced nuclear energy systems: Gas Fast Reactor (GFR), Lead Fast Reactor (LFR), Molten Salt Reactor (MSR), Super-Critical Water Reactor (SCWR), Sodium Fast Reactor (SFR) and Very High Temperature Reactor (VHTR) were selected and related international joint research is being carried out. Since 2007, consultations began between the Proliferation Resistance and Physical Protection Working Group (PRPPWG) and steering committees of the six reactor system to publish a PRPP characteristics white paper, which was published in 2011. The purpose of this white paper is to provide preliminary information on the PR&PP advantages of each system and to suggest directions for optimizing PR&PP evaluation.

The GIF PRPPWG, in collaboration with system designers, has decided to update the white paper on the PR&PP robustness of the GIF design concept and reflect changes to the six reactor system, along with improved unique PR&PP capabilities since the initial publication of the white paper.

The update was conducted using a modified common template, the composition of which includes elements of the PR&PP evaluation methodology, and system elements of the proposed design concept, potential nuclear proliferation and physical protection targets, and the design's response to threats posed by state actors. The revised white paper also includes information on PR-related design features based on IAEA STR-332. Currently, the LFR and SFR white papers have been completed and the remaining 4 white papers are in the final stage.

In this paper, we will analyze the contents of the revised GIF PRPP white papers and provide an overview of an activity that evaluates PRPP of GEN-IV nuclear energy systems.

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Keywords: GEN-IV, Nuclear fuel cycle, Proliferation resistance, Physical protection, Safeguards

Assessment Methodology for Proliferation Scenario of Nuclear Strategic Items by Employing Bayesian Network

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In January 2004, Abdul Qadeer Khan was arrested for black market sale of nuclear technology and equipment to Iran, North Korea, Libya and possibly others. This made the black market for nuclear weapons known as the Khan Network known to the world and revealed that non-state individuals and organizations could purchase nuclear weapons-related programs. As such, the failure of export control of nuclear strategic items can lead to international nuclear proliferation.

Although export control per se has been re-enforced as unraveling Khan's activities, however, there has been no case of quantitatively evaluating the export risk of trigger items and applying them to regulatory guidelines. Therefore, in order to assess the export risk of nuclear strategic items, setup of a reasonable proliferation scenario with probabilistic causal relationship and an efficient method to quantify the export risk of each component of the scenario are required.

In this study, in order to utilize in the export risk assessment, potential proliferation paths and scenarios of the nuclear strategic items were investigated and analyzed. Khan network, which is an actual case of nuclear proliferation, was investigated to derive a quantitative causal relationship that can be used in the evaluation of export risk. In addition, the Freeman network, which studied the path through which nuclear weapons proliferate, and the Freeman-Mella network, which further expanded it, were analyzed.

Bayesian networks were investigated as a methodology for scenario evaluation. A Bayesian network refers to a probabilistic graphic model that expresses a set of variables and their conditional dependencies via a directed acyclic graph and a conditional probability table. Bayesian network is widely used in the field of risk assessment, and efficient algorithms can perform inference and learning in Bayesian networks. In applying the Bayesian network to the export risk assessment of trigger items, an establishment of reasonable proliferation scenario on the Bayesian network is imperative.

As a result, black-market goods and proliferation routes were checked to make nuclear weapons through the Khan Network, and the routes for nuclear weapons development were confirmed through the Freeman Network and the Freeman-Mella Network. Since the Bayesian network can evaluate the uncertainties and probabilistically analyze the causal relationship between factors on the scenario grid, it turns out to be an efficient tool to quantitatively evaluate the export risk of trigger items. The results of this study are expected to be used in the Export Risk Assessment System for trigger items.

Acknowledgements

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Keywords: Proliferation scenarios, Bayesian network, Trigger items, Export risk assessment

Uncertainty Propagation in the Pu Accounting System of Pyroprocessing for the Safeguards Purpose

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The Pu accounting system in pyroprocessing is the most important component to establish the large scale pyroprocessing system in the context of nuclear safeguards. One of the methods suggested to measure Pu in the pyroprocessing system is the Pu-to-Cm-244-ratio method. In order to implement this method, the ratio of Pu and Cm-244 should be correctly and precisely evaluated in the head-end process which is a process before the key-pyroprocessing system such as the electric-reduction, electric-refining, and electric-winning process. During the key-pyroprocessing processes, the Pu mass can be evaluated by the multiplication of the Pu-to-Cm-244-ratio obtained in the head-end process and the Cm-244 mass measured by a neutron detection system. There are various uncertainty sources in this in-direct Pu measurement system. For example, the uncertainty of the Pu-to-Cm-244-ratio can come from the evaluating Pu mass and the Cm-244 mass for samples in the head-end process. Moreover, the uncertainty of the Cm-244 mass during the key-pyroprocessing inherently comes from the neutron detection system. Therefore, we investigated how those uncertainties sources can propagate through the key-pyroprocessing. Then, the available Pu throughput based on the results of the uncertainty propagation was analyzed. For that, we first assumed the detection system to evaluate the Pu-to-Cm-244-ratio in the head-end process and the Cm-244 mass during the key-pyroprocessing. It was assumed that the Pu mass in the samples was evaluated by THE NAME OF DETECTION SYSTEM as the destructive assay method. We assumed that the Cm-244 mass was measured by the NAME OF DETECTION SYSTEM. Because of each parameter has the uncertainty represented by the standard deviation in this study, we conducted the random sampling using the MATLAB software based on all random variables (or parameters) are following the normal distribution. This sampling scheme was iterated 10,000 times. Finally, we could discuss the conceptual available Pu throughput under those Pu measurement system was being applied.

Keywords: Safeguards, Pyroprocess, Pu-to-Cm-244-ratio method, Plutonium, Nuclear material accountancy

Establishment of the Evaluation Criteria for Environmental Swipe Samples From Spent Fuel Handling Facilities

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The International Atomic Energy Agency (IAEA) proposed environmental sampling program to monitor undeclared nuclear activities in member states. The Republic of Korea (ROK), as a member state of the IAEA, has also been operating domestic environmental sampling program. Particle samples from facilities are collected by cotton swiping, screened and analyzed. However, there is no evaluation criteria to verify undeclared nuclear activities in the ROK, especially for hot cells. Therefore, we have established evaluation criteria for the analysis results of cotton swipe samples using MCNP6 and Origen-Arp code. The reference spent fuel rod as a control group has the following characteristics: PLUS7 type assembly (CE 16×16), 4.5wt% enrichment, 45 GWD/MTU discharge burnup and 30 years of cooling after discharge. The spent fuel rod was divided into 200 sections in the axial direction to obtain the average fission energy deposition of neutron for each section using MCNP6 and average discharge burnup for each section was calculated based on the fission energy deposition of neutron. The isotopic ratio of each section was obtained using Origen-Arp. A hypothesis testing was used to select which isotope ratio could be used as a conditional criteria. As a result, we have established the evaluation criteria for environmental swipe samples from domestic hot cells once burnup information is given.

Keywords: Environmental sample, Spent fuel, Evaluation criteria, Burn-up condition

Energy Dependency in Angularly Biased Emission of Prompt Neutrons by ^{244}Cm and ^{240}Pu

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The phenomena that prompt neutrons of fission-correlated tend to be emitted in opposite or same direction have been physically identified. This unique property was found to be dominated by momentum distribution of accelerated fission fragments, and it has been tried to apply that to applications for nuclear safeguards. Previous studies confirmed inversely proportional property of asymmetry from prompt neutron angular distributions according to sample mass of nuclear materials for diverse nuclear isotopes, especially spontaneously fissionable nuclides, such as ^{252}Cf , ^{244}Cm and ^{240}Pu . This correlation presented the possibility that the fission information (e.g., emission angle) can be applicable to estimate neutron multiplication of nuclear material samples. However, for further practical applications, thorough investigation of those properties from diverse perspectives needs to be done. Basically, the angular distributions can be measured via fast neutron detection using organic scintillators without neutron thermalization, so energy threshold is necessarily applied due to high gamma sensitivity. Also, the approach to estimate multiplication via anisotropic emission properties would be limited for such samples whose masses are larger than 1.4–2.0 kg or for several nuclides which indicate very similar asymmetry tendency to sample mass. To that end, we investigated the energy dependency of prompt neutron anisotropic emissions by ^{244}Cm and ^{240}Pu using Monte Carlo simulations. In the results, the asymmetry of prompt neutron angular distributions showed a dependency on neutron energy thresholds, and its characteristic tendency was confirmed for both nuclides.

Acknowledgements

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Keywords: Monte Carlo simulation, Fission neutron anisotropic emission, Energy-dependent asymmetry, Prompt neutron angular distribution, Nuclear identification, Nuclear safeguards

Development of Prototype Code for MUF and Sigma MUF Calculation in Real Bulk Handling Facility

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Material balance evaluation is an important factor in facilities dealing with nuclear materials because it can quantitatively determine whether nuclear material is diverted or not. The evaluation techniques calculating material balance such as MUF (Material Unaccounted For) and sigma MUF are being studied for application to domestic bulk handling facility.

In this study, a prototype code has been developed to benchmark overseas research cases and to perform preliminary evaluation for domestic cases. This code calculates MUF and sigma MUF based on facility information related to safeguards approaches. User can set facility configuration such as Material Balance Area (MBA), Key Measurement Points (KMP), and measurement uncertainties, which are related to material balance evaluation, as an Excel format. Then MUF and sigma MUF are evaluated based on recorded nuclear material amount at input, output, and inventory measurement points during the Material Balance Period (MBP) based on the facility information. The main feature of this code is that it is designed to enable material balance evaluation based on measurement accountancy records in real bulk handling facility. For example, it was considered that nuclear material measurement record format used in a domestic nuclear fuel manufacturing facility could be used in this code as an input file with minimal modification to evaluate MUF and sigma MUF. The other feature is that this code has Graphical User Interface (GUI). Therefore, even users who do not have sufficient knowledge of material balance evaluation technique can easily evaluate MUF and sigma MUF using the code.

Benchmarking study of overseas research cases will be performed to verify the methodology applied in the code. The evaluation algorithm of the code complies with the techniques presented by IAEA. Therefore, it is expected that this code derives the same results as overseas research cases. After the verification study, preliminary evaluation will be also conducted on domestic bulk handling facilities using this code, and through this study, facility uncertainty (sigma MUF) occurring in these domestic facilities will be estimated.

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Keywords: Material balance evaluation, Material unaccounted for, Sigma MUF, Bulk handling facility

1분과

핵주기정책·규제 및 비확산(Poster)

Development of VR Content on Contingency Situation for High Immersion Education Course

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With the Fourth Industrial Revolution, KINAC (Korea Institute of Nuclear Nonproliferation and Control) has developed the Virtual Reality (VR) education system and operated contents based on motion tracking since 2016. Pilot VR content on contingency situation (such as terrorist attacks) was developed in 2017. However, according to rapid development of VR technology over the past five years, the VR education system & contents had to be improved in 2021. This paper describes the features and utilization of newly developed VR content on contingency situation in 2021.

The background of new VR content on contingency situation is the main gate of a nuclear facility, and the main mission for trainees is to protect a nuclear facilities from terrorist attacks as a security personnel. The most important factor considered in the development process were “how realistically contingency situations can be expressed in the content” and “how strongly trainees can be immersed in the content”. Therefore various threats and countermeasures had been needed to reflect into the content to enhance realism. Finally this VR content has been developed including “Threat types and response options”, (such as 2 Response spot, Threat types, Day or night setting, Aiming laser setting, Difficulty setting, etc) that could be combined separately. Therefore the important feature of new content is the function to customize contingency situations. For example, barriers available for trainees to conceal were changed depending on the response spot (A: cars, B: guard posts, barricades, etc.). And trainees also can move freely within a certain range (3–6 m). These features can enhance the immersion level of trainees in content.

The VR-based education provides the main benefit that high-risk training (which is difficult to easily experience in reality) could be provided as a low-risk repetitive training. In this regard, VR education system & contents are the most optimized for practical training related to contingency situation. This VR content will be used for awareness programs and other legal education courses. And it will contribute to enhance physical protection personnel's capacity and strengthen the nuclear security culture.

Keywords: Physical protection, Contingency situation, VR, Education and practice

Nuclear Non-proliferation and Nuclear Security Factors for Evaluating an Export Risk of Trigger Items

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In order to prevent the proliferation of weapons of mass destruction, including nuclear weapons, export controls on strategic items are being enforced under international norms. Strategic items refer to goods that require restrictions such as export permits for international peace and safety and national security in accordance with the principles of the international export control system. Among them, trigger items are required to be specially managed by the Korea Institute of Nuclear Non-proliferation and Control (KINAC) for export licensing of goods and technologies designated by Nuclear Supply Group.

In general, nuclear export control starts with classification that determines whether export items can be used for strategic items. Due to the qualitative characteristics of the export control process, there take place lots of difficulties in each step of the process. Moreover, exports of nuclear material and technologies to various customer countries are expected to phenomenally increase inasmuch as the rapid growth of domestic competences in nuclear technology and nuclear industry. However, there is a high possibility that the current professional judgment of export license depends on the subjective judgment of the examiner as it contains many qualitative contents. In order to cope with the increased burden of classification activities, the KINAC intends to develop efficient and robust export risk assessment system for trigger items. Since the export risk evaluation system requires to process risk information on import countries and strategic items in parallel, risk factors for nuclear non-proliferation and nuclear security of each country need to be collected, properly selected and reasonably prioritized.

In order to establish a nuclear non-proliferation and nuclear security profile for each country, this study investigated international treaties, regimes, actions and internal status of each country related to nuclear non-proliferation and nuclear security. Based on this information, proper risk factors for nuclear non-proliferation and nuclear security were selected and priorities were determined through expert review. Country risk profile, which is to be used in the export risk assessment system for nuclear non-proliferation and nuclear security, was collected and generated.

As a result, a total of 25 risk factors were selected, including 11 international treaties, 4 international regimes, 2 international actions and 8 internal statuses. Priorities were determined from the 1st to the 7th by classifying the selected factors as essential, optional and noted items. These results will be used to evaluate the country risk for the export risk assessment system for nuclear non-proliferation and nuclear security.

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Keywords: Nuclear non-proliferation, Nuclear security, Export risk, Trigger items, Profile

The Soil Sample Collection Scenario to Confirm Nuclear Activity of Denuclearization Core Facility

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The International Atomic Energy Agency recommends using radioisotope ratio signatures to understand core denuclearization facilities' operation history and nuclear activities. By understanding changes in the state of uranium-isolated natural conditions such as $^{235}\text{U}/^{238}\text{U}$ through analysis of collected soil samples, it is possible to determine whether or not the nuclear activity is occurring in the area where the samples were collected.

The soil sample collection scenario to confirm the nuclear activity of the denuclearization core facility considers the topography, surrounding environment, and ease of analysis. Scenarios were constructed on the premise of the sampling situation presented in ISO 18400-104. The collected soil sample must contain sufficient amounts of nuclear material for analysis. Therefore, it is essential to determine the number of samples collected according to the sampling concentration and pattern. The radiochemical laboratory, which is representative of the denuclearization core facility, the location of pollutants such as waste management facilities and spent fuel rod storage can be identified relatively accurately. When considering one facility as an object of interest, applying a regional phase in which samples are collected at sampling concentrations of 1 to 10 points per square kilometer is appropriate. Also, considering that the attached buildings are densely packed, and the surrounding site is almost certain to be contaminated, a stratified random sampling pattern should be applied.

When constructing a scenario using limited information, as in the case of the Democratic People's Republic of Korea, it is necessary to consider the reality and representativeness of the sample. DQO (Data Quality Objective) was utilized to supplement these limitations. DQO is widely used to develop survey plans for contaminated sites during the decommissioning of nuclear power plants. The situation where the area of interest is contaminated with radioactive materials is the same, so it is also appropriate to apply DQO to the scenario.

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Keywords: Nuclear activity, Denuclearization core facility, Scenario development, Soil sampling

A Study on the Criteria for Assessment Major Items on the Nuclear Fuel Cycle

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Methods for evaluating and quantifying nuclear proliferation in the nuclear cycle are established through the IAEA safeguards and Additional Protocols. Export controls are also being implemented based on these items, but items and guidelines are continuously established or revised through the Nuclear Suppliers Group. The most important item for nuclear weapons is, of course, nuclear material. However, facilities and items for producing plutonium or high enriched uranium in a nuclear cycle are also very important.

The NSG guidelines has no priority for each item or evaluation criteria for component parts. The NSG recommends that the transfer of component parts should meet the purpose of export controls, and that the government continue to take appropriate actions as it can to achieve this aim and make practical definitions of component parts. The guideline also suggests to implement catch-all control that can contribute to the proliferation even though it is not subject to be controlled. So, control criteria for these component parts and catch-all items are different and ambiguous for each country. In order to eliminate this ambiguity, priority should be set for each item and evaluation methods for component parts should be established for more accurate and systematic implementation of export controls.

In order to evaluate the major items of the nuclear cycle, first of all, the process weight included in the nuclear cycle is required. Next, the weight of the essential equipment list for safeguard and export control items is calculated. Finally, the evaluation of the item itself is performed. The factors of item evaluation are facility scale predictability, size, material, accessibility to nuclear materials, manufacturing technology, specificity and complexity. Through these three steps, all items can be ranked and evaluated by importance. Regarding component parts, the size and value occupied by the item, whether it is reusable, whether it is a detachable part, etc. will be important evaluation factors. In particular, states or non-states actors that want to develop clandestine nuclear program will try to achieve their goals with uncontrolled items, component parts assembly, and technology rather than finished products.

The importance of these items and the results derived from the development of evaluation methodology can be used as an important criterion for governments to make decisions and will be the important basis for evaluating nuclear trade risks assessment.

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Keywords: Major items, Nuclear suppliers group, Nuclear related items, Components parts

A Study on the Methodology for the Nuclear Trade Risk Assessment

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The International Risk Control Council defines risk as the negative and uncertain consequences of an event or activity related to something of human value. Risk assessment in the field of non-proliferation can be used as a method to quantify the risk of nuclear proliferation when making major decisions. Through this, it can be used in decision making in export control such as export license and situational license.

The nuclear proliferation resistance evaluation started in 1980 led by the IAEA and the U.S. DOE. and was divided into three areas, material characteristics, technical characteristics, and institutional characteristics, and evaluated with different standards and evaluation methods for each of 14 nuclear proliferation resistance evaluation factors. The model for quantitatively evaluating nuclear non-proliferation began in 1984 and has been continuously studied. It can be used as a tool for the international community to derive risk warnings in advance to prevent nuclear proliferation activities. Fuhmann and Gartzke, and others quantified the political, economic, and international situation and analyzed the consequent impact on the development of nuclear weapons. It is divided into three main factors, current capacity, domestic political situation, and international political situation, and each of them is divided into 3 categories and analyzed for a total of 9 parameters. Economic capacity, industrial capacity, nuclear capacity, leader intentions, political structure, public opinion, competition, alliances, international norms, etc. In addition to previous studies, it is necessary to consider time changes. The trade situation changes in real time, and this method can be applied through the Bayesian network or Markov chain methodology. It is expected that more specific and clear nuclear trade risk assessment will be possible if realistically evaluable and simple factors are used in the factors of previous studies and state-of-the-art indicators such as the Nuclear Threat Index or the Peddling Peril Index are used.

The guidelines recommended by the international export control system are the same, but the standards and procedures implemented by government are different. Even the same importing country may be different depending on various variables. Since export controls are based on some degree of variability and there is no nuclear trade risk assessment method so far, South Korea is implementing decisions through qualitative assessment. It is expected that more efficient and effective regulation will be possible if South Korea establishes its own nuclear trade risk assessment methodology and uses it as basic data for decision-making.

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Keywords: Nuclear trade, Risk assessment, Nuclear proliferation, Export license, Bayesian network

A Study on Export Control Methods for the Foreigner Recruitment

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Recently, as various export projects and international joint research are being pursued, the number of recruitment foreigners is increasing for business promotion. As foreign employment increases, various issues related to this are occurring. In this study, it is analyzed issues related to foreign recruitment and try to derive export control methods to solve them.

There are two major export control issues related to the recruitment of foreigners. The first issue is whether export controls are applied to the recruitment foreigners, and the second issue is the technology transferred to hired foreign employees for the purpose of performing their duties.

The first is whether or not export permits are applied to hiring foreigners. The current Foreign Trade Act includes control over the transfer of borders and concepts of nationality. However, hiring can be viewed as an export from a foreign country to Korea in that the knowledge that a person possesses is used by paying a price. Therefore, it is judged that the employment itself cannot be regarded as an object of export license.

The second is to transfer the company's technology to foreign employees in order to perform their duties. Under the current law, if technology is transferred from a local person or a domestic corporation to a foreigner or a foreign corporation, it is considered to be subject to export control. However, in the case of foreigners hired by a domestic corporation, the interpretation is ambiguous because they belong to a domestic corporation. Currently, the Ministry of Trade, Industry and Energy (MOTIE), in charge of dual-use items, interprets the transfer to a foreigner belonging to a domestic corporation as an export license subject, but treats it as an export license exemption.

In the case of trigger-list items, the majority of export licenses are carried out by corporations rather than individuals. In other words, in the case of individuals belonging to a corporation rather than an individual to which they do not belong, it is judged appropriate to assign the export control responsibility to the corporation. In this case, the technology transferred to the hired foreigner is not subject to permission because it is transferred within the same corporation. In addition, even when an employee transfers a corporation's technology to a third country through overseas dispatch or business trip, the corporation can fulfill its export control responsibilities regardless of employee's nationality.

Currently, legal review on export control entities is in progress. If it is possible to judge an individual belonging to a corporation as a corporation, it is expected that the major issues related to export control related to the recruitment of foreigners mentioned above can be resolved.

Keywords: Recruitment, Foreigner, Technology transfer, Individual, Corporation

A Study on Laws Related to the Nuclear Export Control Inspection System for Efficient Technology Export Control

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With the development of information and communication technology, the export control system has been also required to change as the form of export changed from goods to technology. In the past, export control implementations centered on goods were conducted through customs clearance. In the process of customs clearance, practical regulatory activities were possible, such as whether an export license was obtained or not, and an activity to compare the actual product with the product on the permit.

However, in the case of technology transfer through information and communication networks, it is very difficult to implement export control regulations on technology before it is actually transferred because there is no separate customs process like goods. In the end, for technology, it is necessary to issue a license in a lump for a large amount of technology transfer per business unit and to implement export control regulations through follow-up management. For follow-up management, it is essential to inspect whether the conditions set at the time of issuance of the license are fulfilled and whether the technology is transferred outside the scope of the license.

In the case of dual-use items, the Compliance Program (hereinafter referred to as “CP system”) is applied in which companies or organizations that want to export can receive rights from implementation self-classification and export license benefits if they have the necessary system (organization, rule, education system, etc.) for export management. Regulatory institutions periodically inspect companies or institutions implementing the CP system to check whether the CP system is being implemented rightly and the details of technology export, and apply administrative measures such as rating adjustment of the CP system and disqualification according to the inspection results.

However, in the case of trigger-list items, the CP system is not applied, and it is difficult to apply the current CP system tailored to dual-use items as it is. Currently, in the case of trigger-list items, a large amount of technology is licensed for each business unit through the Nuclear Plant Technology Export License System (hereinafter referred to as “plant export license system”). When issuing the plant export license system, regulatory institution guarantees the autonomy of export control implementation and imposes various conditions, and they have implement periodic inspections. Nevertheless, the law does not specify inspection procedures like the CP system, it is difficult to take administrative action on matters found after follow-up management. Therefore, this reduces the practical effect of the plant export license system.

Currently, the study was being prepared for an export license system associated the plant export license system and inspection system. If this license system is enacted, it will be possible to support export of the nuclear industry and to establish an efficient and systematic nuclear technology export control.

Keywords: Nuclear technology, Inspection, Export license, Compliance program, Trigger-List items

A Study on Issues and Solutions for Nuclear Material Export and Import System

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Those who want to import and export nuclear materials such as Source Material and Fissionable Material can import and export only after implementing Advance Notification of Transfer of Nuclear Material (hereinafter referred to as “advance notification system”) and obtaining of Regime on Nuclear Material Export and Import Requirements Confirmation (hereinafter referred to as “requirements confirmation system”). The advance notification system is to receive information from exporters and importers to implement the obligation of ROK government to report to the International Atomic Energy Agency (IAEA) on imports and exports of nuclear materials. The requirements confirmation system is the government approves imports and exports by receiving relevant information from the exporters and importers to check whether the conditions necessary for customs clearance are met.

The first issue is that the export and import of nuclear material is possible only if both systems are implemented. The first issue is that the export and import of nuclear material is possible only if both systems are implemented. The regulatory institution in charge of the advanced notification system and the requirements confirmation system is also the same as the Nuclear Safety and Energy Safety Committee (NSSC), and the contents required for implementation for both systems have no huge difference. In the sense that each of these two systems must be implemented, unnecessary administrative burden is incurred.

The second aspect is the effectiveness of the requirement confirmation system exemption. An exemption conditions are described in the requirements confirmation system guidelines. If the exemption conditions are satisfied, the import and export of nuclear material is possible without approval of the requirements confirmation. However, it can be seen that there is no benefit to the exemption because the standards for the conditions of exemption are not clear, the standards are also set in other regulation, and the documents required for the application for exemption are same with approval. Therefore, it can be seen that there is practically no utility for the exemption.

In order to solve these issues, a study is preparing to amend related laws and regulations and to enact an integrated the advanced notification system so that the requirements confirmation system and advance report can be processed at once. In addition, in order to establish exemption standards suitable for both the regulation in the aspect of nuclear non-proliferation and import and export activities of exporters and importers, a study is underway to analyze the cases of the requirements confirmation system applied so far. In the future, it is planned to utilize the results of this study for efficient nuclear material import and export regulation.

Keywords: Nuclear material, Export, Import, Administrative burden

Integration of Export Control System Databases

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The Korean government controls the exports of strategic items for implementing multilateral export control regimes. Strategic items are equipment, software and technology that can be used for the manufacture, development, use, or storage of conventional weapons and weapons of mass destruction. They are classified into three categories: (1) nuclear-related dual-use items; (2) non-nuclear-related dual-use items; and (3) military items. Especially for nuclear-related equipment and technology, Nuclear Safety and Security Commission takes charge of controlling exports, and Korea Institute of Nuclear Non-proliferation and Control (KINAC) conducts a technical review on export permit.

KINAC operates various types of databases for export control regulatory activities. However, practical use of current data is inefficient owing to the scattered data across multiple databases. Moreover, data that have same attributes are stored in several databases as different databases are used for each regulatory task. This phenomenon leads to the difficulties of consistent data handling, wasting data storage space, and occurrence of data inconsistency among databases due to the different processes of data update.

Data warehouse is a database which specialized in reporting and analyzing data. It collects and stores data accumulated over time in one single place to provide meaningful business insights. While standard operational database is optimized in processing real-time data transactions, it is hard to handle data on itself as other functions of operational database stops during data analysis. This may harm the overall system stability thus the necessity of new repository has emerged.

Data warehouse model improves data quality by sorting and processing existing data, and then it provides better information to the end-user. Whereas traditional database stores data by functional process, data warehouse integrates and stores data by subject from a user's point of view to support decision making. In the case of export control regulation, there are vast amounts of data which change over time such as strategic items, denial lists, and agreements between countries. Therefore, it is difficult for regulators to immediately obtain data necessary for decision making, including export permit. Based on the features of data warehouse, we suggest applying data warehouse model to the database integration is the best method to settle such issues.

In this paper, we introduced integration method for export control databases. We expect regulators to be able to perform on export control regulation work more effectively with subject-oriented data produced by such methodology.

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Keywords: Export control, Database, Data warehouse, Nuclear non-proliferation

Investigation on the ^{238}U -daughter-isotope-ratio Changes in the Uranium Mill Tailings for the Purpose of Safeguards

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In nature, the radioactivity of U-238 (half-life: 4.51E9 years) is equal to the activity of U-238's daughter isotopes such as Th-234 (half-life: 24.1 days), Pa-234 (half-life: 1.17 minutes), U-234 (half-life: 2.47E5 years), Ra-226 (half-life: 1,602 years), Rn-222 (half-life: 3.82 days), etc. This phenomenon is known as the 'secular equilibrium' which can be observed when the half-life of a mother isotope (U-238) is much longer than that of a daughter isotope (Th-234). Before the milling process, the radioactivity ratio of U-238 to each daughter isotope is approximately one. However, after milling, the most of uranium isotopes (more than 90%) are separated from other isotopes and come out as the product material. The mill tailings include most of the daughter isotopes and the rest of the uranium (less than 10%). Because separating of U-238 from uranium ores in the milling process, the radioactivity ratio between U-238 and its daughters in the tailings has varied as a function of time. Moreover, we investigate what kind of isotope ratio can be a signature to inform when that the tailings are produced. For that, the analytical solutions of Bate equations for U-238 to Rn-222 are obtained. To provide the milling process effect, we intentionally remove 90% of uranium isotopes. We finally observe the Th-234 and Pa-234 activity changes as a function of time. Moreover, the variation of ratios for Th-234-to-Rn-222 and Ra-226-to-Th-234 as a function of time within one year are shown in the figures.

In conclusion, those ratios are applicable to analyze when the tailings are produced within one year. Based on this viewpoint, we could suggest the application of those ratio for the safeguards purpose. For example, the tailings should be produced based on the operation record. However, once operators produce unannounced material, the unexpected tailings can be produced. In this case, analyzing the ratios of Th-234-to-Rn-222 and Ra-226-to-Th-234 in the unexpected tailings can be a signature to inform inspectors that when the milling process was clandestinely operated.

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Keywords: Safeguards, Tailings, Uranium decay chain, Bate equation, Secular equilibrium

Investigation on the Difference Between the Proliferation Resistance Evaluation and the Safeguardability Evaluation

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The nuclear non-proliferation issue becomes more critical issues as increasing nuclear facilities and amount of bulk material demand in the world. Therefore, there are various methods developed to evaluate the proliferation resistance (i.e. International Project on Innovative Nuclear Reactors and Fuel Cycle (INPRO), Proliferation Resistance and Physical Protection (PR&PP), Multi-Attribute Utility Analysis (MAUA), Technological Opportunities to increase the Proliferation resistance of global civilian nuclear power Systems methodology (TOPS), Simplified Approach for PR Assessment of nuclear systems (SAPRA), Risk Informed Proliferation Analysis (RIPA)) and safeguardability (Acquisition Pathway Analysis, Facility Safeguardability Analysis (FSA), High Reliability Safeguards Method (HRS), Safeguardability check list). Even though the both terminologies are significant in the context of nuclear non-proliferation, there are some difference between two. In this study, we briefly discussed the difference between the proliferation resistance evaluation and the safeguardability evaluation. In this study, the INPRO method and the FSA method were selected as a representative method for the proliferation resistance evaluation and the safeguardability evaluation respectively. Then, the raw data for the spent fuel composition produced by Korea Atomic Energy Research Institute were utilized. The major similarity between the two is 1) they both considered Pu and U materials, 2) they intended to prohibit nuclear diversion, 3) Containment & Surveillance systems are important in the two methods, 4) they covered entire nuclear fuel cycle facilities, and 5) the acquisition path, the design information, and the cost are critical parameter to measure the proliferation resistance and safeguardability. The main differences are 1) the proliferation resistance considered isotope compositions (i.e. Pu-238, Pu-239, Pu-240, etc), however, the safeguardability considered the amount of total plutonium (i.e. 1SQ of Pu is 8 kg). The attractiveness including heat, activity, and toxicity was significant variables in terms of the proliferation resistance evaluation. In contrast, those were less important to measure the safeguardability. The safeguardability measure is more close to the regulation works than the proliferation resistance measure, because the safeguardability linked to the safeguards agreement later.

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Keywords: Safeguardability, Proliferation resistance, INPRO, Facility safeguardability analysis, Safeguards

A Study on the Need of Revising Article 107 of the Nuclear Safety Act on Export and Import Procedures

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In order to prevent the spread of nuclear weapons, the international community controls the export and import of nuclear-related materials, equipment, and technologies through international treaties and agreements related to nuclear non-proliferation regime based on the NPT. Korea is also obligated to manage exporting and importing nuclear-related materials, equipment and technologies by joining the NPT and signing the Safeguards agreement with the IAEA and nuclear cooperation agreements with other countries.

In accordance with these international treaties, Korea imposes such obligations on companies and the people who import and export nuclear items through Article 107 of the Nuclear Safety Act and Article 151 of the Enforcement Decree of the same Act. However, Article 107 of the Nuclear Safety Act was created in the past when nuclear items were mainly imported, including only regulatory obligations and not specific procedures and methods. Currently, Korea is handling the obligations under nuclear international treaties through the export permit system for strategic goods in accordance with the Foreign Trade Act or through official documents between the government and importers and exporters. In addition, there is no basis for punishment if the import/export procedure is intentionally not followed.

Now that nuclear power plants are exported, import and export obligations under nuclear international treaties should be more actively fulfilled. Article 107 of the Nuclear Safety Act needs to be completely revised, to transit from the regulatory obligations to import and export licenses. The revised contents should include clear processing standards and procedures to make it easier to find and recognize import and export obligations under nuclear international treaties. The basis for the establishment and operation of an online system for efficient nuclear export and import control management is also required. In addition, it is necessary to clearly present the targets and standards for punishment in case of violation of duty and to link with the Korea Customs Service to prevent customs clearance if import/export licenses are not obtained. The details of the full amendment will be carried out in a future study.

Keywords: Non-proliferation, Export and import procedures, Nuclear safety act, International treaty, Nuclear cooperation agreement

A Study on the Confirmation Procedure for Items Subject to the New ROK-U.S. Nuclear Cooperation Agreement

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In April 2015, the government of the Republic of Korea and the United States of America signed a new Nuclear Cooperation Agreement (NCA). Subsequently, in April 2016, the Nuclear Safety and Security Commission (NSSC) of the ROK and the Department of Energy (DOE) of the U.S. signed the Administrative Agreement (AA) pursuant to the new Nuclear Cooperation Agreement. It is important to clearly define the Criteria and Confirmation procedure of items subject to the NCA because the NCA and the AA impose various obligations such as import and export control procedures and annual inventory reports.

According to Article 2 of the new NCA, nuclear materials, decelerators, equipment and components, whether directly or through a third country, are subject to the agreement upon their entry into the territorial jurisdiction of the receiving country when the exporting country notifies the importing country of the intended transfer and the importing country confirms the receipt of such notification in writing. The AA is intended to stipulate details such as document form and notification procedure for implementing the nuclear cooperation agreement.

The important point here is to exchange transfer information between the government of exporting country and the government of importing country. Therefore, only items notified from the exporting country in accordance with the import and export procedures of the AA should be subject to the new NCA. It is not only physically impossible for the government of the importing country to check whether the items imported by the importer are items of the new NCA without implementing AA procedures, but also wastes unnecessary administrative power. Notifying the items subject to the agreement is intended to ensure that the exporting country has control over the export items, so it is essential and efficient for the exporting country to check and notify when issuing an export license. For importing countries, it is appropriate that items imported from exporters by domestic companies without implementing administrative agreement procedures are excluded from the items subject to the agreement.

Currently, Korea notifies domestic companies of the items subject to the agreement through an official letter for each individual case, and it may be necessary to revise the laws to establish the specific procedure for notification of the agreement items. The specific procedure for the government to notify import and export companies of items subject to the agreement will be conducted in a future study.

Keywords: Nuclear cooperation agreement, Administrative agreement, Export, Import, Annual reports, Prior notification

A Study on Current Status of Obligated Items Subject to Bilateral Nuclear Cooperation Agreements

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The bilateral nuclear cooperation agreement provides the basis for technical and economic exchanges and R&D cooperation in the nuclear industry, while also having an important role in the peaceful use of nuclear power.

The government of Republic of Korea has signed nuclear cooperation agreements with 29 countries, starting with signing the agreement with the United States of America in 1956. In particular, to specify detailed procedures necessary for the implementation of the agreement, supplementary and additional arrangements with the United States, Canada, and Australia have been prepared and entered into force.

For the effective implementation and cooperation of the agreements, an accurate understanding of the obligated items subject to the agreement by country is required. In addition, since problems that may occur in the actual field require understanding the characteristics of the domestic nuclear industry as well as the categories of items subject to agreement, various reviews of current system are required for efficient and effective implementation.

This paper summarizes the items and obligations subject to each country's agreement and analyzes the domestic status of possession and handling process. Also, focusing on the United States, Canada, Australia, and Japan, obligated items such as nuclear materials, materials and equipment, components, and technologies subject to the agreements are organized and analyzed.

As a result, it is intended to derive the specificity of domestic item management system by checking the current status and management process of major nuclear licensees. It can also help establish a practical management system for items subject to bilateral nuclear cooperation agreements in Korea.

Keywords: Bilateral nuclear cooperation agreement, Obligated items subject to the agreement, International transfer, Export and import control

Development of Material Balance Evaluation Methodology Based on Measurement Uncertainty in GUM

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The implementation of safeguards about nuclear material accountancy is required based on the agreements between International Atomic Energy Agency (IAEA) and member States, and from 1970s, IAEA developed and applied a methodology to estimate measurement error for the material balance evaluation (MBE). The measurement error or uncertainty is essential for MBE since the estimated material unaccounted for (MUF) can be the criterion for determining whether nuclear materials are diverted or not. In the IAEA methodology, the traditional error components, random and systematic error, are dealt with for the estimation of measurement errors from the top-down.

In contrast, since the publication of the ‘Guide to the Expression of Uncertainty in Measurement’ (GUM), safeguards laboratories, one of the pillars of the nuclear material accountancy, has begun to apply this international metrological standard to their measurement results. GUM-based measurement uncertainty estimation is a bottom-up methodology in the consideration of all factors affect the measurement uncertainty.

In this study, a general methodology for material balance evaluation in use of regulating bulk handling facilities (BHF) in Korea is derived from the IAEA safeguards methodology and GUM, and measurement uncertainties for each stratum in a specific BHF are propagated and estimated based on GUM-based measurement uncertainty. With these uncertainties, the MUF of the BHF is calculated and for the identification of its statistical meaning, several tests with estimated MUF are also conducted to verify the null hypothesis that nuclear materials are diverted.

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Keywords: Material balance evaluation (MBE), Nuclear material accountancy, Safeguards, GUM, Material unaccounted for (MUF), Bulk handling facility (BHF)

Validation Analysis of X-ray Scan Images Through Nuclear Related Item Models

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The container cargo inspection detector is a non-intrusive inspection device that can inspect internal goods without removing the cargo from the container and widely placed in airports and ports. The cargo inspection detector uses 6 MeV or more high-energy X-rays of to penetrate a metal thickness of 3,000 mm to detect dangerous goods and smuggled items in containers. It is appropriate to apply a container cargo inspection detector when inspecting containers which include nuclear-related items, but it is difficult to use large scale detector at a laboratory scale experiment because of its bulky size and costly price.

Therefore, the experiment has taken by the carry-on baggage X-ray scanner with 50 KeV energy generally used in general security screening and ten nuclear-related items made of metal which are downsized or small sized will be scanned by low energy X-ray scanner.

All the experimental data will be analyzed by theoretical data and when the obtained X-ray image is corrected through the mass attenuation coefficient curve as X-ray energy increases, it is analyzed whether the data is valid by comparing it with the X-ray image obtained through the container cargo inspection detector.

The X-ray images gained through this experiment and analysis will be used as basic source data for supervised learning when developing an algorithm to detect strategic items by deep learning technology.

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Keywords: X-ray image, X-ray scanner, Container cargo inspection, Nuclear related item

A Review of Safety Evaluation Methodology for Application to Physical Protection Regulation Considering Radiation Effect

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In order to manage spent nuclear fuel at the national level, it is necessary to develop key methodology and relevant technologies for physical protection as well as safety, and to effectively and efficiently link related policies, regulations, and technologies.

In light of international recommendations for physical protection, it is necessary to rationalize protection requirements and prepare regulatory guideline or standards so that protection measures are taken in consideration of radiation effects and safety factors, which are lacking in current physical protection measures. Therefore, in this study, the relationship between the existing safety evaluation was reviewed in order to develop a methodology and regulation tools for evaluating physical protection performance and radiation impact from sabotage for nuclear material during transportation and temporary storage. In order to review the relationship between safety evaluation and risk assessment of the structural conditions for spent nuclear fuel transport cask and storage pool, major legal technical guideline and requirements are summarized, and the evaluation factors under accident conditions are reviewed in consideration of the utility from the point of view of physical protection.

The results of this study will be used to evaluate whether new physical protection regulations required in the future can be satisfied by using the existing safety evaluation methodology, or whether additional evaluation ranges or methods are required.

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Keywords: Spent fuel, Physical protection, Transport cask, Storage pool, Radiation effect

Measurement of Uranium Enrichment Using Laser-Induced Breakdown Spectroscopy

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The uranium enrichment is important in safeguards and nuclear forensics. Mass spectrometry and gamma-ray spectroscopy are widely used for the measurement of uranium enrichment. Laser-Induced Breakdown Spectroscopy (LIBS) can be an option for the measurement of uranium enrichment. Compared to the mass spectrometry, the measurement time of uranium enrichment using LIBS was shorter. Correction in gamma-ray spectroscopy, which is necessary due to the large difference in gamma-ray energies emitted from uranium isotopes, is not necessary in LIBS for the measurement of the uranium enrichment. In the present work, the measurement of uranium enrichment is performed using LIBS combined with Principal Component Analysis (PCA).

Laser pulses from a Q-switched Nd:YAG laser with wavelength of 532 nm, and energy around 10 mJ were focussed on a sample surface, and plasma emission from the sample surface was measured using Echelle spectrometer and EMCCD. All the measurements were performed in argon with atmospheric pressure. The uranium enrichments of the samples used in the present work were 1.6–19.8%, and the numbers of LIBS data at each uranium enrichment was 5–10. U II 424.437 nm lines with isotopic shift of –25.1 pm was selected in the present work. Although the ^{235}U and ^{238}U lines were not completely resolved due to the limited spectral resolution of our detector instrument, the intensity of the ^{235}U position in the LIBS spectrum tended to increase as the uranium enrichment of the samples was higher. PCA was performed using commercial software (Unscrambler X, CAMO), and most of the data with the same uranium enrichment were divided into separated groups in PCA result.

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Keywords: Uranium enrichment, Safeguards, Nuclear forensics, LIBS, PCA

Analysis of Safeguards Status for Domestic Bulk Handling Facility and Overseas Inspection Support Software

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Korea has established an independent regulatory system separately from the IAEA and has been conducting national inspections since 2015. However, since the criteria, the statistical verification system and the detailed procedures for bulk handling facility are insufficient, there is a limit to confirming the adequacy of the nuclear material accounting system for nuclear facilities.

In order to establish a systematic national inspection, process characteristics analysis, sampling plan and field inspection activities should be derived according to the facility. Therefore, it is necessary to develop an on-site national inspection support software that can support various verification activities during national inspections to improve the efficiency of national inspections for bulk handling facility.

This paper describes the status of safeguards implementation for KEPCO NF, a representative bulk handling facility in Korea, and the various overseas inspection support software such as CIOSP, VARO and PIE-IS to develop on-site national inspectional support software suitable for domestic facilities. The necessary functions for the development of the national inspection support software were derived, and it is necessary to develop the software in a modular way for each function, so that it is easy to delete, add and modify major function.

Keywords: Safeguards, Bulk handling facility, National inspection, Inspection support software

Comparison of Machine Learning Algorithms for Pulse Shape Discrimination in Neutron/Gamma Scintillation Detectors

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Spent fuels emit a lot of neutrons and gamma-rays of various energies, which depend on isotopes of transuranic elements and fission products. For the safeguards purpose, separate detection of neutrons and gamma-rays is required to monitor special nuclear materials (SNM) in nuclear facilities where both radiations coexist. Organic scintillators such as BC-501A and EJ-276, and Li doped inorganic scintillators such as NaIL, CLLB, and CLYC for neutron and gamma-ray measurement are recently commercialized. And various pulse shape discrimination (PSD) methods using signal processing DAQs enable neutron and gamma-ray events to be distinguishable. Nowadays, machine learning (ML) algorithms are used in radiation measurement and analysis such as spectrum analysis and pile-up correction. ML-based PSD methods can be substitutive for the conventional PSD methods.

In this study, several ML algorithms are applied to distinguish neutron and gamma-ray signals, such as Support Vector Machine (SVM) which is a conventional machine learning algorithm, and Convolutional Neural Network (CNN) which is a deep learning algorithm. To obtain a training data set, waveforms of a CLLB scintillation detector with neutron and gamma-ray isotopes were measured with a CAEN DT5730 digitizer, which has 500 MS/sec sampling rate and 14 bit resolution. The obtained waveforms were normalized to remove the energy feature which may confuse the artificial intelligence. The normalized waveform data in time domain were directly used to train the models which extracts the inherent feature of neutron and gamma-ray waveforms. To investigate time and frequency domain effects in the machine learning model and compare the models with time and frequency domain, normalized waveform data in time domain were converted into frequency (Fourier and/or Wavelet) domain and used for input data of the ML models. After the model validation, the performance of the SVM and CNN with time and frequency domain were evaluated in terms of neutron and gamma-ray pulse shape discrimination. Additionally, a conventional PSD with signal processing was compared with the ML based PSD methods.

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Keywords: Machine learning, Neural network, NDA, Pulse shape discrimination (PSD)

Investigation on Advanced Intrusion Detection and Surveillance System for Physical Protection in Nuclear Facilities

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The highly advanced technologies using artificial intelligence (AI) and big data have been developed and applied for various industry field. They have been also combined with security system to operate the system efficiently such as the CCTV with AI technology and access control with big data analysis. In specific, as demands for improving security capabilities are increasing in nuclear facilities, some nuclear power plants are trying to operate new technologies. In this study, the technologies, which have been mainly examined in the security field, were reviewed in terms of the physical protection in nuclear security: (1) intelligent video surveillance (2) light detection and ranging (LiDAR) (3) biometrics.

First of all, the intelligent video surveillance is one of the prominent technologies in the security system. It is the advanced technology adding deep learning technology based on object and behavior recognition analysis. In traditionally, the security guards should continuously monitor the CCTV screen to identify suspicious behavior. As the intelligent video surveillance, which is the system with AI machine learning algorithm, is introduced to physical protection system, it could supplement the work of security staffs and increase the efficiency and accuracy of security system.

LiDAR technology identifies distance, size, direction near the sensor by using laser light pulses. It sends out pulses and read the reflection bouncing back from the objects. As LiDAR has a 360-degree detection range and generate 3D images, it can monitor the protected area with high accuracy. In this point of view, the security system including LiDAR is considered as the prominent technology for intrusion detection by complementing the drawback of the traditional security system such as limited detection range, sensitive to weather or lighting conditions.

Biometrics is commonly used in security system for access control to identify individuals whether they are allowed in or not. Types of authentication include fingerprints, finger geometry, face and iris recognition, vein recognition. Due to the COVID-19, the non-contact biometric technologies have been increasingly developed and improved such as face and iris recognition. However, the confidence and accuracy of new technologies are relatively low compared to fingerprints measurement which has been used widely so far. Therefore, it is necessary to identify the vulnerabilities to avoid any possible error and build countermeasures against malicious attacks.

In this study, three promising technologies were examined in terms of physical protection in nuclear security and it would appear that it is essential to identify some factors prior to introducing the technologies: what are the vulnerabilities, how to respond them and what is required to prepare the countermeasures. The results from further investigation is anticipated to significantly contribute to the national physical protection.

Keywords: Physical protection, Nuclear security, Intelligent video surveillance, LiDAR, Biometrics

A Study on the Need for Institutional Improvement According to the Status of KAERI's Export and Import of Nuclear Material

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Export controls are necessary to prevent international peace and countries of concern or terrorist organizations from acquiring materials, equipments, parts and technologies necessary for the development and Weapons of Mass Destruction (WMD). International community has to configure the international export control regimes work together for this purpose. Republic of Korea (ROK) is a member of the international export control regimes and reflects the export control guidelines in domestic laws and complies with them. This paper will deal with the difficulties of the application process for Export and Import of nuclear materials submitted to the Korea Atomic Energy Research Institute (KAERI) and the need for institutional improvement.

KAERI is applying for export control of Trigger list Items to the Nuclear Safety and Security Commission (NSSC). At this time, Trigger list items are reviewed by Korea Institute of Nuclear Nonproliferation And Control (KINAC) examination using the Nuclear Export and imPort System (NEPS) that examines and judges trigger list, and finally receives approval from NSSC. The Export and Import of nuclear material must apply for an international transfer Advance Notification in common. In accordance with the safeguards agreement and additional protocols concluded with the International Atomic Energy Agency (IAEA), it is required to report on nuclear material transferred domestically and internationally.

The Export and Import of nuclear material must apply for an international transfer Advance Notification in common. The Advance Notification must be submitted at least 2 weeks in advance, depending on the amount of nuclear material. After Advance Notification, Nuclear Material Requirements are applied through NEPS. Nuclear Material Import Approval result must be issued prior to Export and Import because it is a necessary document in the customs. Advance Notification and the Nuclear Material Import Approval are linked in NEPS. In this process, it is difficult to repeatedly apply for changes in Advance Notification and Nuclear Material Import Approval.

Currently, the average number of applications for Export and Import of nuclear materials within KAERI is around 10 cases per year, but the processing time is increasing due to continuous change application. If the application is possible change after Export and Import, it is expected that repetitive work can be reduced and efficient work. In the future, it is need to continuous understanding Korea Customs system and related laws and regulations.

Keywords: Strategic items, Export and import, Nuclear materials, Advanced notification, Nuclear material import approval, Trigger list

Challenges in Managing Spent Fuel and Radioactive Waste From Nuclear Power Plant Decommissioning in Germany - Focusing on the 7th National Report of Joint Convention on the Safety Spent Fuel Management and on the Safety of Radioactive Waste Management

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After the promulgation of the 1st Act on the Closure of Nuclear Power Plants (Act on the Structured Phase-out of the Utilisation of Nuclear Energy for the Commercial Generation of Electricity) in 2002, in the aftermath of the Fukushima nuclear accident, Germany proceeded with the 13th amendment to the current Atomic Energy Act and adopted the Atomausstieg (Nuclear Phase out).

Therefore by 2022, all nuclear power plants in Germany will be permanently shut down. Germany shifted from deferred dismantling to immediate dismantling, considering the cost aspect, ethical aspect of not transferring responsibility to the next generation, and difficulties in dismantling by human chain disconnection that may occur in the situation of delayed dismantling, which is consistent with the IAEA's recommendations.

If the nuclear facility is dismantled immediately, the amount of radioactive waste is generated at once, so it is necessary to ensure that sufficient funds are established for radioactive waste disposal and appropriate disposal sites are secured.

Based on Germany's 7th national report on the 'Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management', an international agreement, this study aims to review and present challenges for gradual dismantling of nuclear power plants, including financial resources and manpower for safe dismantling, Spent fuel and operation, and dismantling radioactive waste disposal.

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Keywords: Joint convention, Permanent shutdown, Immediate dismantling, 7th national report, Disposal site, Financial resource

A Study on Effective Post-License Management for Conditional Export License on Trigger List-Items

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The export license system has been established to prevent indiscriminate transfers of goods, software, and technology (Trigger List-Items) necessary for the development, production, and use of nuclear weapons in accordance with the Foreign Trade Act. If continuous managements are required even after issuance of an export license, additional license conditions are imposed to stipulate follow-up obligations to export licensees.

A transfer of 'technology' requires continuous management rather than a transfer of 'goods'. Export licenses for the 'technology' are usually issued in a contract unit, not individual units when a transfer scale is large and the list to be transferred are uncertain. In this case, a post-license management is required more strictly as details of each transfer are not clearly defined at the time of license issuance. The License on Technology Export of Nuclear Plant is especially imposed on post-license obligations such as post strategic items confirmation and transfer report, leading to the necessity of comprehensive post-license management.

However, it is difficult to comprehensively manage the conditional license as the current Nuclear Export Control System (NEPS) has no post-license functions. A Licenser sends an official document to notify license conditions to a licensee and individually confirms whether the condition is fulfilled whenever deadline approaches. This implantation is prone to omissions and brings about unnecessary administrative loads to both licensor and licensee.

Also, if the licensor or licensee is changed, tracing history data is difficult. In addition, the post strategic items confirmation has been being implemented through classification function in current NEPS due to absence of post-license functions. It needs to be managed separately as the evaluation criteria, application subject and application deadline could be different from those of the classification, according to the post-license management guidelines.

This study suggests that an additional system should be established to manage post-license obligations for conditional export licenses. The system should have comprehensive functions to implement post-license obligations and related policy such as post strategic items confirmation, transfer report, export control training certificate submission and emergency technology transfer application. An automatic function notifying the periodic obligation whenever deadline comes will also be helpful. The result of this study will be utilized for the construction of post-license management system after reflecting the additional opinions from licensors and licensees.

Keywords: Trigger list, Export license, Technology transfer, Nuclear export control system (NEPS)

A Study on Risk Assessment of Accidentally Acquired Nuclear Material

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The Nuclear Non Proliferation Treaty (NPT) is the cornerstone of the global nonproliferation regime. The NPT parties shall promise and fulfill their obligations in the areas of non-proliferation, disarmament and peaceful use of nuclear materials and related technologies. To fulfill its commitments under NPT, South Korea has signed a binding protection contract with the International Atomic Energy Agency (IAEA) to apply IAEA protection measures in Korea. Under the safeguards agreement, South Korea is obliged to report to the IAEA on all nuclear materials and specific types of nuclear energy research and manufacturing, and accept IAEA inspections in the region. And South Korea is obliged by the IAEA to ensure that the Safeguards are applied to all nuclear materials in South Korea to ensure that all nuclear materials in the country are not diverted for non-peaceful purposes. Old laboratories and sites dealing with nuclear materials have late detection of nuclear materials used long ago when the concept of safeguards measures was not secured. This has become an issue recently.

Therefore, the risk assessment of accidental acquired nuclear material was studied. It is necessary to conduct voluntary and systematic safety management, rather than passively responding to an incident such as the accidental acquisition of nuclear material. In the case of chemical plants overseas, through quantitative risk assessment (QRA), process risk recognition, risk assessment, risk determination, and risk reduction measures are established and implemented. In order to prepare an effective accident prevention and emergency response plan, it is necessary to calculate a reasonable quantitative risk and determine the risk acceptance level.

Keywords: Safeguards, Inspection, Accidental gain, Nuclear material, Risk assessment

Analysis of Evaluation Parameters for Safeguardability

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Safeguardability is defined as the degree of ease with which a system can be effectively and efficiently safeguarded. Safeguards by Design (SBD) is the process of including international safeguards considerations throughout all phases of a nuclear facility life time. For introducing a new nuclear facility to the State, the IAEA recommends SBD to ensure the effective and efficient implementation of safeguards at facility for the State and the IAEA. Therefore, evaluation of safeguardability can be applied to implement and optimize SBD for new nuclear facility. Safeguardability is one of the proliferation resistance (PR) attributes. For PR evaluation, various methodologies have been studied and TOPS, INPRO, and GIF PRPP are representative methodologies developed as an international cooperation. PR is that characteristic of a nuclear system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States in order to acquire nuclear weapons or other nuclear explosive devices. Therefore, safeguardability evaluation and PR evaluation have different objective and it is necessary to develop the evaluation parameters specialized for safeguardability.

In this study, evaluation parameters for safeguardability were analyzed to estimate the achievement of IAEA safeguards objective, applicability of safeguards measures, and ease of IAEA inspection. These parameters were investigated based on the evaluation parameters of TOPS, INPRO, and GIF PRPP in which international PR and safeguards experts participated and IAEA safeguards manual.

Acknowledgements

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Keywords: Safeguardability, Evaluation parameters, Safeguards by design (SBD), Proliferation resistance

Classification and Management Procedures for Disused/Residual Nuclear Materials in KAERI

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The Korea Atomic Energy Research Institute consists of 11 MBA and 39 KMPs. The types of uranium and nuclear materials used by each facility are diverse, and each facility is conducting experiments and research according to its purpose. If an experiment, research, or related project is completed, nuclear material that can no longer be used (disused) or is not used (residual) may occur.

Therefore, KAERI classifies these nuclear materials as disused/residual nuclear materials and stores and manages them. Disused nuclear material in liquid and solid form means unirradiated nuclear material that is ended its use or is going to discard. Residual nuclear materials means unirradiated nuclear material that has been retained or has ceased to be used in laboratories. Residual nuclear material exists only in solid form, and if there is a request for transfer from another department, it can be handed over according to the regulations and procedures for nuclear material accountancy and control.

In principle, it is essential to maintain accountancy data for all nuclear materials, including the disused/residual materials.

In the case of disused nuclear material in liquid form, the department with the material must obtain a certification report by chemical analysis and handed over to the disposal department in accordance with the procedures related to radioactive liquid waste treatment. At the same time, the safeguards department reports the material to the IAEA as a retained waste and excludes it from the safeguards subject. If there is a plan for solidification of the liquid, the safeguards department notifies the plan to the IAEA and applies for the termination of safeguards of the disused nuclear material in liquid form. Thereafter, it is terminated from the subject of safeguards according to IAEA approval. The Solidified material is transferred and stored in Uranium Residue Storage Facility.

In the case of residual nuclear material, it is stored and managed in the Nuclear Material Storage Facility that can be distributed the material. Residual material can be reused when there is a request for transfer from another department.

Keywords: Disused/Residual nuclear material, IAEA, Safeguards, Retained waste

A Review of Safeguards Status of Korea Atomic Energy Research Institute

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In 1975, the Republic of Korea signed the Safeguards Agreements and Subsidiary Arrangements. In accordance with the agreements, Facility Attachment of TRIGA II and III, the first nuclear facilities, is signed in 1976, and the IAEA conducted inspections on TRIGA II and III. In 1978, with the operation of the CANDU Fuel Fabrication Plant (CFFP), the IAEA conducted inspection activities for KAERI in earnest.

In Korea, AP was finally applied in 2004 through the revision of the Atomic Energy Act and the ratification of the National Assembly. In August 2004, the KAERI submitted the first expanded declaration, and IAEA conducted Complementary Accesses (CA).

As transparency in nuclear activities has been secured, in June 2008, Korea received a broader conclusion that all nuclear materials were under peaceful activities from IAEA and Integrated Safeguards (IS) is applied. Consequently, subject facilities to IAEA safeguards were classified into Category I and II. The frequency of IAEA inspections was reduced to half that of the existing safeguards system, but the RII, which was notified two hours ago, was accepted.

In 2016, the IAEA reviewed the application of the State-Level safeguards Approach (SLA), a new concept of safeguards implementation, which evaluated the capacity of the state as a whole. Accordingly, subject facilities to IAEA safeguards were classified into Category I, II, and III. The frequency of inspections is determined by the IAEA based on the factors of each state, but member states are unknown because this part is confidential. The IAEA conducts PIVs based on annual operation schedule submitted by the facility. In principle, notice is given one week in advance for PIV and two hours in advance for RII.

Since 2016, the average number of inspections for PIV and RII during 5 years was 10.7 and the number of inspections including additional approach inspections was 13.7. As of the end of September 2021, the number of inspections in PIV and RII was 14, and the number of inspections including CAs was 20, which increased compared to the average (6 CAs, 7 PIVs, 7 RIIs and 1 TV in 2021). Comparing the number of inspections during the same period in 2020, it is confirmed that the number of inspections this year has increased (the number of inspections in PIV and RII was 12, and the number of inspections including CAs was 17).

It is thought that there will be effects such as simplified procedures, self-isolation exemption of inspectors upon arrival in Korea, and exclusion from the countries where the corona mutant virus occurred. Furthermore, it is not certain, but it seems that there will be an impact related to the application of the upgraded SLA.

As mentioned above, although the number of inspections increased due to unexpected circumstances, but KAERI did all the IAEA inspections without the failure and proved that nuclear transparency in nuclear activities of the facility was secured.

Keywords: IAEA safeguards, State level safeguards approach, PIV, RII

Simulation Procedure for the Evaluation of Physical Protection System Effectiveness Using AVERT Software

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Nuclear facilities handling radioactive materials have been regarded as the attractive target of bombing, sabotage, and theft for malicious purposes. To oppose the capabilities of such potential adversaries, an effective physical protection system (PPS) with the functions of detection, delay, and response is necessary, and the total time for detection, delay, and response must be less than the time for the adversaries to complete their attack goals. Thus, force-on-force exercises have been commonly conducted to evaluate the effectiveness of the PPS while it requires a lot of cost, time, and manpower. To minimize such limitations through computer simulations, the automated vulnerability evaluation for risks of terrorism (AVERT) software has been widely used in the field of nuclear security. In this research, the basic procedures considered for the AVERT simulation were set and summarized. Then, it was analyzed that what type of simulation results are achieved and how these results are adopted to evaluate the effectiveness of the PPS. In particular, the basic procedures were composed of considerations and assumptions, methods of information gathering, modeling procedures for geometry and component, determination of input parameter values, set-up of intrusion scenarios, and revision of logical contradictions. Based on the above-mentioned procedures, simulation results can be variously obtained such as vulnerable paths to intrusion, the probabilities of detections, interruptions, and neutralizations as a function of time, the number of neutralizations per the number of intrusions, etc. Eventually, those data are used to determine the intrusion paths during force-on-force exercises and to evaluate and enhance the effectiveness of the PPS by analyzing the vulnerable paths to intrusion and taking measures.

Keywords: Nuclear facility, Physical protection system, AVERT simulation, Vulnerability assessment

The ROK's Efforts to Achieve IAEA Safeguards Implementation Goals in Pandemic

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The World Health Organization (WHO) declared COVID-19 a pandemic on March 11. Through this, economic damage has increased worldwide, and daily life itself has changed significantly, including policy changes and movement restrictions in many countries. This naturally led to changes in IAEA safeguards activities and SSACs in IAEA member countries. In this situation, the ROK has been striving from various perspectives to successfully conduct IAEA safeguards activities. In the early days of the pandemic, there were various difficult situations such as global travel restrictions and entry restrictions under COVID-19, but the Korean government has been working closely with the IAEA for SG verification activities from the early stages of the pandemic to the present. As a result, a total of 104 SGs activities (95 inspections and 9 CAs) could be successfully performed.

In particular, in order to achieve the IAEA safeguards goals, we adjusted the operation schedule of domestic nuclear facilities and cooperated closely with IAEA to ensure that IAEA inspections are carried out in a timely manner. In addition, the ROK provided information package to IAEA inspectors to enable timely access to nuclear facilities of the ROK. It includes the Korean government's entry procedures, infectious disease prevention and control measures, facility access procedures, and emergency contacts. Based on these results, it was an important opportunity to reconsider how important it is to implement safeguards activities even in situations where close cooperation between IAEA, NSSC and nuclear facilities is difficult.

Keywords: COVID-19, Safeguards implementation, IAEA, Information package

Analysis of Drone Regulatory System Trends in Korea to Establish Protection Systems Against the Threat of Unmanned Aerial Vehicles on Nuclear Facilities

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As the global drone industry develops, the use of drones both for hobby and commercial use has increased. The status of ultra-light vehicle operator qualifications from 2004 to June 2021, provided by the Korea Transportation Safety Authority, shows the number of ultra-light vehicle qualification rapidly increasing from 2017. Before the pilot qualification standards tightened in 2020, most of the pilots counted were pilots for drones weighing more than 12 kg for business use. The actual number of drone users will be much higher. As the number of drones operating increases, illegal drone flight and terrorism cases are also in concern. The nuclear power plant is a no-fly zone around an 18 km radius, so unauthorized flights are not allowed. However, the number of drone appearances from 2016 to 2021 is steadily increasing, and unconfirmed numbers is thought to be much higher. Nuclear power plants, which are classified as “A” level national facilities, need to consider the adaptation of drone protection systems by the change of social trends. Prior to adopt drone protection systems for nuclear facilities, it is necessary to summarize major laws regulating drones and major changes after the ‘Drone Law’ was legislated in 2020.

In this study, we investigate and organize the laws for managing and regulating the scope of drones currently stipulated in the new ‘Drone Law’ legislated in 2020. We also investigated the latest drone terrorism and threats around the world and identified considerations for the adaptation of drone protection facilities to nuclear facilities.

Keywords: UAV, Nuclear facilities, Physical protection, UAS, Drone

2분과

사용후핵연료 처분전관리 (Oral)

Criticality Safety Analysis on Dry Storage Module (DSM) With PWR Spent Nuclear Fuel

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Basically, DSM should be designed to maintain subcriticality in consideration of normal, ab-normal and accident conditions for criticality safety. The canister model in DSM is 24 FA container, which consists of a basket cell, a neutron absorber, and a neutron absorber cover. For criticality evaluation, the SCALE6.1.3 program that developed by ORNL was used and the library of ENDF 238 group was applied. Validation of SCALE6.1.3 program was carried out in accordance with the NUREG-6698 methodology based on 236 Benchmark Experiments presented by the OECD/NEA. As a result of validation of SCALE codes, bias and uncertainty were assessed as 0.00661 and 0.01079 respectively. And the maximum k-value (k_{eff}) for normal conditions at the 95% probability and 95% confidence level is 0.94153 which is below the design basis of 0.95. As an accident condition, DSM was considered in the full flood condition as the most conservative assumption. The technical standards for criticality evaluation are in accordance with US 10CFR72.124 "Criteria for Nuclear Criticality Safety" and American Atomic Energy Society regulatory guidelines ANSI/ANS-57.9. The criticality technical standards applied to the DSM design are as follows.

- The effective neutron multiplication factor, k_{eff} , including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident-level conditions.

- At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident level conditions would need to occur before an accidental criticality is deemed to be possible (i.e., double contingency principle).

- When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period. The neutron-absorbing materials' continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of these materials cannot occur over the life of the facility.

- Criticality safety of the cask system should not rely on credit for more than 75 percent of the neutron poison material in fixed neutron absorbers when subject to standard acceptance tests. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.

Keywords: Criticality, k_{eff} , Effective neutron multiplication factor, SCALE

Evaluation of Heat Transfer Performance as a Function of Canister Internal Pressure in SNF Dry Storage Module

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The storage of spent nuclear fuel (SNF) is very important for the stable operation of a nuclear power plant. In Korea, PWR SNF has stored in spent fuel pool of Nuclear Power Plants (NPPs) and various studies are being conducted for dry storage system. KHNP is developing a dry storage module for dry storage of PWR SNF. Since the dry storage module stores several canisters at the same time, it is very important to remove the heat inside the canisters.

In this study, the heat transfer performance as a function of internal pressure filled with the helium gas inside the canister was evaluated. For thermal evaluation of canister, Fluent 19.2, a commercial computational fluid dynamics (CFD) program, was used. The RNG k-epsilon turbulence model was used for the air flow region, and the density of Helium was input to change according to the temperature to simulate natural convection. The inside of the canister was used by calculating the effective thermal conductivity. For efficient CFD analysis, the flow analysis domain used 1/4 model for normal condition and 1/2 model for off-normal and accident conditions. The pressure of the filling gas inside the canister was varied from 1 to 4 atm, and the axial temperature and the maximum cladding temperature at each pressure were evaluated.

As a result of the evaluation, it was found that the dry storage module had sufficient heat transfer performance when the pressure was higher than 4 atmospheres. Therefore, maintaining the internal pressure of canister above 4 atm is critical to satisfy the thermal design criteria under all analysis conditions to maintain the integrity of cladding and canister components.

Keywords: Spent nuclear fuel, Canister, Dry storage, Thermal evaluation

Review of the Handling Procedure for the Spent Fuel Sealed Container at the PIEF

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The Post-Irradiation Examination Facility (PIEF) of Korea Atomic Energy Research Institute (KAERI) is a facility operated for the purpose of evaluating the integrity of nuclear fuels, including support for R&D, performance evaluation, and identification of the cause of damage. For these purpose, a total of 1699 rods of spent nuclear fuels were transported several times and safely stored in a storage pool within the facility. Various destructive and non-destructive tests were performed on some of these fuels, and are currently being stored in the form of specimens and fragments. However, in 2016, the National Assembly, local government, and citizens expressed concern about the fact that KAERI, located in the large city, is storing spent nuclear fuels. Accordingly, KAERI decided to lay the foundation for transporting spent nuclear fuels to power plants within 5 years. Therefore, a sealed container of Westinghouse called Quiver was introduced to more safely store and transport the intact and damaged spent nuclear fuels, and the Quivers and related handling tools were designed and manufactured to suit the environment of PIEF. The procedures for handling these Quivers and handling tools in the storage pool of PIEF were arranged, and whether there were any parts for improvement to make handling safer or easier was reviewed.

Keywords: Westinghouse quiver, Spent nuclear fuel management, Post irradiation examination facility, Sealed container

A Comparative Study on the Cladding Failure Criteria of a High Burnup Spent Nuclear Fuel Rod

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The cladding of Spent Nuclear Fuel (SNF) should be protected from external loads or any degradation mechanisms which can cause gross rupture of fuel rod during the dry storage periods. With this regard, the mechanical integrity of SNF rods and cladding are evaluated using numerical methods as a part of safety assessment of dry storage systems and operations. In this evaluation, one critical element is the proper setting of the cladding failure criterion. The failure criterion should be obtained by well-designed experiments with irradiated fuel cladding specimens. However, those experiments are technically demanding and there exist very limited data available in literature. The failure criterion should be set considering the failure mode of the fuel rod, and for plastic rupture caused by excessive external loads the plastic strain is often used as proper failure criterion. According to the NUREG-1864 report, the typical fracture plastic strain that can cause failure of the cladding was expected to be in the range of 1.0–4.0%. However, the data are obtained with uniaxial or biaxial tension test with defueled fuel cladding specimen and can hardly be applied to the evaluation of actual fuel rod failure where very complicated 3-d stress state arise in the vicinity of pellet and clad interfaces. In this study, the failure criterion of fuel cladding which can account for the deflection, through thickness stress development are introduced and compared.

The first failure criterion of the cladding is the plastic bending strain of 1% computed from the centerline curvature of the fuel rod. This failure criterion is determined as the maximum major structural strain of the cladding compared to the strain critical point related to the material properties of the cladding during dynamic impact. The second cladding failure criterion is proposed in this work which can account for the localized stress concentration and the through-thickness stress development. The membrane plus bending stress of the von Mises stress was used for the failure criterion, and the value is 749 MPa for the reference fuel rod considered in this work.

For comparison of the failure criteria, detailed finite element models are constructed, one with fully bonded fuel-clad interface, the other with fully de-bonded pellet-clad interface. In the static analyses considering pure bending condition, the two criteria provide similar results, with smaller critical loads for the fully de-bonded fuel rod. However, in dynamic impact simulations, they showed different trend. The stress-based criterion predicted earlier failure of fully bonded fuel rod than the fully de-bonded fuel rod. This seems mainly due to the large shear stress development in the interface of bonded pellet-cladding. For the fully de-bonded fuel rods, the shear stress cannot develop in this interface and the rod deflects more due to smaller shear resistance. The validity of those failure criteria should be checked with experimental data obtained by actual fuel rod specimen, which is not defueled.

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Keywords: Spent nuclear fuel, High burnup fuel, Simplified model, Failure criterion, Pellet-clad interfacial bonding condition, Finite element analysis

Calibration of Simplified Beam Model Properties of High Burnup Spent Nuclear Fuel Rods Considering Linearized Stress Failure Criterion

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It is essential that the Spent Nuclear Fuel (SNF) maintains a handleable condition without damage to the nuclear fuel rods before the final disposal or reprocessing. In this study, a new failure criterion for SNF Cladding was introduced. When a fuel rod is subject to bending, a very complicated 3-D stress state is generated within the vicinity of the pellet-pellet interface. If the failure strain criterion obtained from the uniaxial tension test or biaxial tube test is applied, very early failure is predicted at the beginning stage of loading with premature through-thickness stress or strain development. The localized contact stress or strain is self-limiting and is not a good candidate for the cladding failure criterion. Therefore, a new cladding failure criterion is proposed in this work which can account for the localized stress concentration and the through-thickness stress development. The failure of the cladding is determined by the membrane plus bending stress generated through the thickness of the cladding which can be calculated by a process called stress linearization along the stress classification line. The membrane plus bending stress of the von Mises stress was used for the failure criterion, and the value is 749 MPa for the reference fuel rod considered in this work.

The material properties of the simplified beam model of a SNF rod were calibrated for a drop accident evaluation by considering the pellet-clad interaction (PCI) of the high burnup fuel rod. In a horizontal drop, which is the most damaging during a drop accident of SNF, the stress in the cladding caused by the inertia action of the pellets has a great effect on the integrity of the fuel rod. The material of the simplified model was modeled as an elastic-plastic material de-scribed with the same equation used for Zircaloy-4 of the cladding material. For the calibration of the properties of the simplified beam model, ISGITH, the optimization program, was used to optimize the calibration parameters to minimize the discrepancy in the behavior of the simplified model and the detailed model.

The simplified beam models were developed based on parameter calibration using design optimization. The purpose of these models is to predict the failure of fuel rods in cask level analyses. To account for the effect of the PCMI, two separate models were developed for the fully bonded pellet-clad interface condition and the fully de-bonded interface condition. Because the beam models cannot simulate the concentrated stress state, stress correction factors were derived which can be used in the prediction of the maximum membrane plus bending stress in the simplified models. The applicability of the simplified models and the stress correction factors were verified in the dynamic impact analyses considering various drop heights.

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Keywords: Spent nuclear fuel, High burnup fuel, Calibration, Membrane plus bending stress, Stress correction factor, Pellet-clad interfacial bonding condition

Optimization of Scintillation Efficiency of Safeguards Equipment for CANDU SNF

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In this study, computer simulation using Light Tools was used to optimize the most efficient size of the P-Terphenyl scintillator to be applied to the next-generation heavy water fuel safety measures equipment. Existing Optical Fiber Probe System (OFPS) uses lithium-glass scintillators, but it was difficult to measure radiation efficiently due to its small light field and long decay time. To solve this problem, the lithium glass scintillator will be replaced with a P-Terphenyl scintillator that has relatively good performance and is easy to obtain. The P-Terphenyl scintillator is an organic crystal scintillator without hygroscopicity and can be used in wet interim storage facilities. In addition, this organic scintillator has a decay time of 3.3 ns and a light yield of 40,000 photons/MeV, which has significantly higher luminous properties than lithium glass scintillators. Moreover, P-Terphenyl scintillators, which can be easily obtained compared to lithium glass scintillators produced in small quantities by custom production and can be prepared in a large area and cut into desired sizes, have an advantage in terms of economy.

Photon transfer efficiency according to the size of the scintillator was calculated to apply a P-Terphenyl scintillator optimized for next-generation safety measures equipment. The shape of the scintillator is a rectangular parallelepiped shape to facilitate production and replacement, and the surface in contact with the optical fiber for the transfer of photons produced in the scintillator is designed to be square. Accordingly, the length of the scintillator and the area in contact with the optical fiber were set as variables. Assuming that all six sides of the scintillator are mirror-polished, the scintillator emission spectrum was set to generate the same wavelength as the P-Terphenyl emission spectrum, and was computerized under the condition that 11,000 photons/mm³ were generated at random locations inside the scintillator. The virtual detector that counts the finally reaching photons is designed to be located at the 10 mm end of the optical fiber in contact with the scintillator.

Through computational simulation, photons generated from 24 sizes of scintillators were transferred to optical fibers. According to the computer simulation result, a scintillator with a width of 4 mm and a length of 10 mm showed the highest efficiency, and a scintillator with a width of 2 mm and a length of 1 mm showed the lowest efficiency. As the volume of the scintillator increases, the number of photons generated in the scintillator increases linearly, but the number of photons transferred to the fixed optical fiber does not increase linearly.

The results of this study will be applied directly to the next generation of water-repellent nuclear fuel safety measures equipment for spent nuclear fuel safety measures. In addition, the developed safeguards equipment will review its performance through the verification of KINAC, and the IAEA certification process will be carried out so that it can be used as an official IAEA safeguards equipment.

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Keywords: Safeguards, CANDU, Spent nuclear fuel, Scintillators, P-terphenyl

Design Criteria of Chloride Induced Stress Corrosion Cracking Tester for Evaluating the Integrity of Spent Fuel Storage Canisters

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One of options in dry storage of spent nuclear fuel is to store in a canister using metal or concrete cask near the coastal area. The storage canister is generally made of austenitic stainless-steel and has the role of an important barrier to encapsulate spent fuels and radioactive materials. The decay heat of the spent fuel is also removed through the canister wall by air cooling. The canister has several welding lines in the wall and lid, which probably have high residual tensile stresses after welding procedure. Interaction between austenitic stainless steel and chloride environment from a sea forms a detrimental condition causing chloride induced stress corrosion cracking (CISCC) in the canister. It is well known that austenitic stainless steels are susceptible to stress corrosion cracking (SCC) in certain circumstances under tensile stress. SCC generally shows an inter-granular cracking or a trans-granular cracking depending on the material condition and temperature.

One challenge to the interim storage of spent fuel in the dry storage canisters awaiting repository disposal is their susceptibility to chloride-induced stress corrosion cracking (CISCC). The canisters placed near coastal regions are subject to salt deposits that can build up with time on the canister surface. These salts can deliquesce resulting in conditions for CISCC. Sea salts accumulates and condenses on the canister surface over a long period of 40 to 100 years. Thus, the corrosion integrity of canisters in the concrete casks is very important because the canister serves to maintain sealability for long period. The development of methodology for evaluating the lifetime of the canisters is also required to assess the corrosion susceptibility with regard to CISCC.

KAERI commences to investigate the crack growth behavior for the evaluation of the corrosion integrity in the canister under chloride atmosphere and secure a CISCC tester this year. This paper presents a study by KAERI to investigate design criteria for the development of the CISCC tester. The design criteria of a CISCC tester are as follows. The CISCC tester consists of dual chamber with the environmental chamber of a constant oven and the constant humidity test cell. Constant displacement is applied using bolt tightening and compact tension test [C(T)] or three-point bending specimens is immersed in chloride solution or injected with a salt steam mist to make an effective chloride environment. The load as a function of time is measured using a load cell or strain gauge. Crack growth rate is measured with the direct current potential drop (DCDP) method and on the surface of fractured C(T) specimen. Stress intensity factor, K is calculated applying the load and crack length to ASTM standard equation and is compared with finite element analysis.

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Keywords: Spent nuclear fuel canister, Austenitic stainless steel, Chloride induced stress corrosion cracking tester, Design criteria, Stress intensity factor

Preliminary Design of a Vacuum Drying Test Equipment for SNF Dry Storage

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Dry storage of Spent Nuclear Fuel (SNF) is an adequate option to secure the storage capacity before the construction of permanent disposal facility. SNF stored in pool must be dried for the storage in Dry storage system. Commercialized drying technologies for maintaining the integrity of SNF and storage structures are vacuum drying process and forced gas dehydration. We are conduct a preliminary design of vacuum drying test equipment to perform vacuum drying performance tests. Vacuum drying is a method of removing moisture from the dry storage cask by lowering the pressure. Moisture removal is assessed by isolating the container from the vacuum system and observing the change in pressure over a prescribed internal of time. An acceptable degree of dryness is maintaining a constant pressure ≤ 3 torr for ≥ 30 minutes. We have designed a vacuum drying test equipment consisting of vacuum pump, vacuum pipe, valves, vacuum gauge, and cold trap. A vacuum pump to reduce the pressure in storage cask and a vacuum gauge for measuring the pressure are important components in the test equipment. Regarding the dryness criteria described in NUREG-1536, we selected a rotary vane vacuum pump that can be used in the rough vacuum range of 3 torr. The internal pressure of Dry storage cask may vary depending on the pumping speed of vacuum pump and the combination of the vacuum pump. Two types of vacuum gauges have been suggested, both directly and indirectly. We had selected a Pirani vacuum gauge which can be measured from atmospheric pressure to medium vacuum among the thermal transfer type gauges. In order to increase the exhaust performance of pump, we had considered the roughing line connecting pump and storage cask, thick and short, are preferable.

Prior to the detailed design for manufacturing test equipment, this paper summarizes the preliminary design results necessary to exert functions suitable for the test purpose.

Acknowledgements

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Keywords: Dry storage, Drying technology, Vacuum drying process, Vacuum pump, Vacuum gauge

2분과

사용후핵연료 처분전관리 (Poster)



A Comparison of Methodologies for Confinement Evaluation of Spent Fuel Dry Storage Systems

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As the capacity of spent fuel storage pools at reactor sites becomes saturated, it is inevitable to introduce spent fuel dry storage systems in Korea. In order to construct and operate the dry storage systems, it is essential to confirm the safety of the systems through safety analysis. Confinement evaluation is an important process of evaluating the doses of nearby public due to radioactive materials leaked from the dry storage cask. Unlike transport systems, confinement evaluation results can be different according to the evaluation methodology.

There are two available confinement evaluation methodologies for normal and off-normal situation. The first methodology from NUREG-1536 is to calculate the atmospheric dispersion factors using the Regulatory Guide 1.145 and to calculate the dose by inhalation and exposure to air submersion. The dose is compared with the annual dose criteria in 10CFR72.104. The second methodology is to calculate the atmospheric dispersion factor using the meteorological data and XOQDOQ code and to calculate the dose using GASPARD code in which the exposure pathway includes external exposure to ground surface and ingestion of agricultural products. The calculated dose is compared with the annual dose criteria to protect people's health and the environment in Korea Nuclear Safety Act.

In order to compare the dose from the two methodologies, we set up the hypothetical storage system. A storage cask of the system contains 21 PWR spent fuel assemblies with an initial enrichment of 4.5wt%, burnup of 45,000 MWD/MTU, and a cooling time of 30 years. During the off-normal condition, it is assumed that the cask is leaked at $1.0 \times 10^{-5} \text{ cm}^3 \cdot \text{sec}^{-1}$. The distance between the cask and the boundary of system is assumed as 700 m and the dispersion and deposition factor from the Kori site are used for the second methodology. The critical dose using the first methodology was 0.02 mSv which is 7.9% of the criteria (0.25 mSv) and the critical dose using the second methodology was 0.008 mSv which is 5.3% of the criteria (0.15 mSv). Although the dispersion factor in the first methodology is more than 10 times higher than that in the second methodology, the reason why the dose is only 2.5 times higher is expected to be that the exposure pathways of the second methodology are more diverse. In this study, it is confirmed that there may be a large difference in dose results according to the confinement evaluation methodology and accordingly, discussions between evaluators and regulators is needed.

Acknowledgements

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Keywords: Confinement evaluation, Spent Fuel, Dry storage, Atmospheric dispersion, Radiation dose, Leakage rate

Structural Safety Analysis on Dry Storage Module (DSM) With PWR Spent Nuclear Fuel

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The SF DSM is a reinforced concrete structure in which the storage cylinder is fixed to the upper slab, and the SF is stored in a dry environment. The DSM must protect the SF to maintain its integrity from the external environment while the SF is stored. In the SF DSM, liner structures are fixed to the upper slab, and storage cylinders are fixed to the liner structures. The canister model in DSM is 24 FA container, which consists of a basket cell, a neutron absorber cover.

For the structural analysis and structural member design of DSM, ANSYS Mechanical, a general-purpose computer program based on finite element analysis, was used. The boundary and constraint conditions used in the DSM analysis model were simulated by reflecting the actual conditions. The lower part of the storage cylinder is constraint support structure, therefore, the boundary condition constrained only two horizontal displacements, and the connection part between the storage cylinder and the module upper slab was constrained by the bond condition. Symmetric boundary conditions were applied to the 1/2 section of the canister, and constraints were applied to the bottom and lid parts according to the analysis conditions. For the canister lid and shell to be welded (joined), full constraint conditions were applied between elements.

In this paper, the structural soundness of the SF DSM was evaluated each condition that may occur in normal, abnormal and accidental conditions as load combination of KEPIC SNC and NUREG-2215 and load combination was performed. For the canister, analysis was also performed under normal, abnormal and accidental conditions, respectively and the load combination results were compared with the stress limit stipulated in the ASME B&PV Code.

Keywords: SF (spent fuel), DSM (dry storage module), Structural analysis, Load combination

Behavior of Sr and Ba of Simulated Oxide Fuel in Molten LiCl Salt

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Safe management of spent nuclear fuel is an obligation and a task for countries that use nuclear power. Since radioactive isotopes that generate a lot of heat from spent nuclear fuel require a large disposal volume, it is very advantageous for disposal or recycling if they are separated from the fuel. Sr-90 and Cs-137 are the main elements that dissipate heat in the early stage of spent fuel generation. One of the methods for separating radioactive isotopes from spent nuclear fuel is to use molten salt. When the spent fuel is immersed in LiCl molten salt at 650°C, SrO, Cs₂O, and BaO react with the salt to dissolve in the salt and separate. In this presentation, the results of studies on the behavior of Sr and Ba of simulated oxide fuel in LiCl salts conducted by KAERI are summarized. Since SrO and BaO in the fuel can be dissolved only when they come into contact with a salt, the degree of dissolution is different depending on the particle size or density of the fuel. When using porous pellets, ICP analysis revealed that about 20–50% and 50–60% of Sr and Ba contained in the fuel at the beginning were dissolved into salts. The amounts of Sr and Ba remaining in the fuel were also quantitatively analyzed.

Keywords: Spent nuclear fuel, Radioactive isotopes, Sr-90, Molten salt, Separation

Re-fabrication of U-10wt%Zr-RE Fuel Slugs Recycling Metallic Fuel Scraps With No Surface Treatment

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The fabrication technology of metallic fuel slugs is the key technology in the fabrication processes of the metallic fuel for Gen-IV nuclear reactors. The injection casting is one of the processes that fulfills these needs, and it has strong advantages in terms of fabricating small diameter castings with a high L/D ratio and a randomly oriented grain structure, enabling precision castings. The metallic fuel scraps during the injection casting of U-10wt%Zr-RE fuel slugs are formed with a weight fraction of about 40%. Moreover, the fuel scraps are classified as a radioactive waste and are stored in temporary storage without recycling. It is necessary to develop a recycling process technology for the scrap wastes in order to reduce the radioactive wastes of the fuel scraps and improve the fabrication yield of the fuel slugs.

In this study, the metallic fuel scraps were recycled with as-cast state without surface removal treatment. The U-10wt%Zr-RE fuel slugs were re-fabricated recycling the fuel scraps such as melt residue and fuel slug butt with as-cast state, resulting in no fuel loss during recycling process. And then, they were characterized to compare with the fuel slugs re-fabricated with surface treatment on the contaminated impurity layer, in order to evaluate the feasibility in the recycling of the fuel scraps.

An additive recycling process is important to reduce the radioactive wastes of the metallic fuel scraps and improve the fabrication yield of metallic fuel. U-10wt%Zr-RE fuel slugs were soundly re-fabricated to the full length of the quartz mold by using the fuel scraps with a modified injection casting. The total impurity content of oxygen, carbon, nitrogen, and silicon was less than 2,000 ppm for the recycled metallic fuel slugs. This study demonstrated the feasibility of the additive fabrication process of fuel slug scraps by re-fabricating the fuel slugs, even in as-cast state without surface removal treatment.

Keywords: Sodium-cooled fast reactor (SFR), Metallic fuel slug, Scrap recycling, Surface impurity layer, As-cast state

Chlorination of Sr in SrZrO₃ With MgCl₂ Molten Salt at 800°C

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Sr-90 is one of the high-decay-heat dissipating nuclides in spent nuclear fuel and needs to be separated from the fuel and separately stored for efficient and effective management of the spent fuel. It is known that Sr can form a solid solution with UO_x fuel matrix or precipitate in grain boundary as an oxide such as SrZrO₃, which is called grey phases, depending on the burnup history and location in the fuel. In our previous study, we tested LiCl and CaCl₂ as reaction media for studying the chemical behavior of SrZrO₃ in these salts at 650°C and 800°C, respectively. It is confirmed that SrZrO₃ is mostly converted into CaZrO₃ and Sr in SrZrO₃ is chlorinated to SrCl₂ after 100 h in CaCl₂ molten salt indicating CaCl₂ could be a good reactant salt but enough time is needed. In this study, MgCl₂ is tested for separating Sr in SrZrO₃ to find a more effective reaction agent at 800°C. Reaction time and the molar ratio of salt to sample were utilized as a variable and structure of the reaction product was analyzed with XRD and morphological analysis was done with SEM-EDS. Sr concentration in the reaction product was confirmed with ICP-OES. It is verified that MgCl₂ is a more efficient reaction media than LiCl and CaCl₂ and MgO, ZrO₂, and SrCl₂ were produced after the reaction, indicating that MgCl₂ can effectively chlorinate SrZrO₃ in a short time at 800°C.

Keywords: Chlorination, Strontium, Molten salt, Spent nuclear fuel, Nuclide separation

Preparation of CaO Beads for Tc and Se Capture

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An accumulation of spent nuclear fuel (SNF) has brought a considerable interest due to its energy and environmental issue, and as an alternative, a geological disposal can be considered to reduce the accumulated SNF. When the fission products (FPs) were effectively separated and removed from SNF, the required area for the disposal will be significantly reduced. In order to separate FPs, the UO₂ will be oxidized to UO₃ through a thermal treatment and various off-gases will be released from the fuels. Among these off-gases, technetium (Tc-99) and selenium (Se-79) should be strictly captured due to their long-lived properties. In this study, cylindrical CaO beads with a diameter of ~2 mm and height of 4–6 mm were fabricated to effectively adsorb oxide forms of Tc and Se. The characteristics of the fabricated CaO beads were examined and the adsorption performance was evaluated.

Keywords: Fission product, Technetium, Selenium, Adsorption, CaO

Analysis on the Domestic and Foreign Status of Spent Nuclear Fuel Storage

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In Korea, spent nuclear fuel are temporarily stored at the reactor site due to absence of interim storage facility. However, all of domestic storage capacity at the reactor site are expected to be saturated by 2065 starting with the Wolseong site in March 2022. Therefore, it is necessary to research for preparing alternatives about this problem suitable for the domestic condition based on the analysis of the domestic and foreign status of spent nuclear fuel storage. In this study, we investigated and analyzed the storage method status of the US, UK, France, and Korea.

Storage methods of spent nuclear fuel can be classified according to storage site and cooling method. In the case of storage methods according to storage site, storage methods can be divided into the AR/RS (At Reactor, Reactor Site), AFR/RS (Away From Reactor, Reactor Site) and AFR/OS (Away From Reactor, Off Site) method. In the case of storage method according to cooling method, storage methods can be divided in to the wet and dry storage. And the dry storage methods of SNF are usually classified into vault, silo, cask. The US uses all of AR/RS, AFR/RS, AFR/OS method. The UK, France and Sweden use AR/RS and AFR/OS method. Korea uses AR/RS and AFR/RS method. France adopts only wet storage method as the spent nuclear fuel storage method. The other countries are using both the wet and dry storage method. The US uses the most various methods as dry storage method. As dry storage method, vault, silo, and cask are used in US. The UK uses vault and cask as dry storage method. In Korea, the Wolseong nuclear power plant uses a wet storage method and a dry storage method. Wolseong nuclear power plant uses vault and silo as dry storage method. In addition, other power plants in Korea use a wet storage method.

In this study, we investigated and analyzed the status of spent nuclear fuel storage method of the US, UK, France, and Korea. It was found that the France uses wet storage method, and other countries use wet and dry storage method. The results of this study can be used as basis when designing spent nuclear fuel storage facilities suitable for the domestic condition.

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Keywords: Spent nuclear fuel, Storage Status, Storage Facility, Wet Storage Method, Dry Storage Method

Estimating the License Fee for the Design Approval of Spent Fuel Storage Cask

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Recently, the Nuclear safety act and its subordinates were amended and introduced the design approval of spent fuel storage cask. According to the amended nuclear safety act, nuclear enterpriser can apply and get the design approval of storage cask but only the operator who operates the nuclear power plant or radioactive waste management facilities can apply the inspection for the fabrication of storage cask.

The aim of this study is to develop a new method of calculating the license fee for the design approval of the spent fuel storage cask. According to the existing notice of the Nuclear Safety and Security Commission (notice no. 2020-04), the unit price and calculation method are different in accordance with the nuclear power plant operator and those other than the nuclear power plant operator.

In the case of the nuclear power plant operator, it is already decided to 2 man·year without the characteristics of the spent fuel storage cask. Whereas in the case of those other than the nuclear power plant operator intend to apply the design approval, the license fee was calculated by applying the standard unit price of the Korea Institute of Nuclear Safety, which is announced every year, according to the workload in compliance with the weight of the storage cask, similar to the case of spent fuel transport cask.

It was designed to calculate 3 man·year based on 50 tons of storage cask and increase 0.5 man·year per 50 tons. In the case of dual-purpose cask, the license fee equivalent to 1.2 man·year is added to the license fee of transport cask.

By estimating the workload on the storage cask and dual purpose cask, the license fee for the design approval of spent fuel storage cask is lower than that of the transport cask, to those other than the nuclear power plant operator.

Keywords: License fee, Spent fuel storage cask, Dual-purpose cask, Man-year, Workload

Electrochemical Characteristic of Zirconium Oxide With UV Irradiation at PWR Primary Water Conditions

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The effects of UV irradiation on electrochemical behavior of oxide films on Zircaloy-4 have been investigated in-situ in this study. The objective is to investigate the effect of Cerenkov radiation on the electrochemical response of fuel cladding materials in reducing high-temperature water conditions. The electrochemical behavior of Zircaloy-4 under UV is investigated with open circuit potential (OCP) and electrochemical impedance spectroscopy (EIS). At room temperature, significant OCP drop is observed when UV is irradiated on the pre-oxidized samples, representing n-type oxide behavior. Also, the oxide impedance of pre-oxidized Zircaloy-4 decreases under UV irradiation, likely due to a reduction of the oxide capacitance. Under in-situ conditions, the temperature and pressure are controlled as 320°C and 15 MPa, and Zircaloy-4 sample is irradiated with a 200 mW·cm⁻² UV source through a sapphire window while in the autoclave. The electrochemical data including oxide resistivity and thickness for irradiated UV and non-irradiated UV conditions are analyzed in the pre-transition regime, to better characterize the effects of UV on oxide electrochemical and transport properties. As reported previously, SEM and TEM characterizations show that magnetite crystals nucleate and grow on top of zirconium oxide under UV irradiation, and the kinetics of zirconium oxide growth are reduced under UV irradiation. A photo-electrochemical mechanism involving Fe photo-deposition and oxide photo-dissolution is presented and further discussed with the new insights of in-situ electrochemical data.

Keywords: UV irradiation, Zirconium oxide, High temperature corrosion, EIS

The Storage Capacity of Spent Fuel in HANARO

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IN this paper describes that how much time will be taken to be filled to capacity at the spent fuel storage. It analyzes the reason why the spent fuel storage is not full of spent fuels even though the HANARO has been operated over 20 years and how much time is left until full the storage. So, it will be useful to predict how to manage the spent fuels in future.

HANARO started operation in 1995 and achieved 26 years of operation and 100 periods. There are several necessary factors for continued operation of HANARO, one of among them is the capacity of spent fuels. According to the HANARO Operating Technical Specifications ‘The spent fuel storage has 24 modules for 36 rods fuel assemblies (each module can be stored 25 assemblies), and 18 modules for 18 rods fuel assemblies (each module can be stored 24 assemblies). During normal operation, three 36-assemblies and two 18-assemblies are replaced per month, and considering the replacement period, it is expected that core reconfiguration will take place about 10 times a year. Therefore, the fuel assembly has a storage capacity of about 20 years.’ It has been operated for over 20 years from now, according to the design specification, the spent fuel storage capacity might be full of spent fuels. However, it has not been full of spent fuels, even though the HANARO has been operated for over 20 years. Currently, the amount of spent fuel in storage is 327 fuels for 36-assemblies and 190 fuels for 18-assemblies. The total storage capacity of 36-assemblies is 600 (currently stock about 65.4%) and 18-assemblies is 432 (currently stock about 44%).

The reason why don’t full the storage is simple. HANARO don’t operate 200 periods. However, how much time is left until full the storage is complication due to changes in fuel and operating period is less than HANARO Operating Technical Specifications a year. In this paper consider many things to influence to occur spent fuel.

Keywords: HANARO, Spent fuel, Storage, Capacity

Investigation of Degradation Modes of Concrete Structures in Dry Cask Storage System

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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 of the storage facility may deteriorate if the interim storage facility operates for more than several decades. Even if deterioration occurs, the concrete structures must maintain its unique functions (shielding 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. This paper investigates the main concrete degradation modes that can occur in dry storage facilities. The possible degradation modes can be classified into two types. 1) When rebar corrosion occurs due to chloride damage or neutralization of concrete, and the concrete is damaged as the corroded rebar expands 2) Deterioration of concrete itself caused by alkali-silica reaction, etc. This paper also shows the concrete deterioration examples observed in the existing spent fuel storage facility and a few examples that evaluated the concrete deterioration and its impact on accident conditions. In the future, we plan to conduct a material test to get the material properties of the degraded concrete, and study how to evaluate the behavior with an FE analysis model.

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Keywords: Dry cask storage system, Interim storage facility, Concrete deterioration, Concrete structures

Surrogate Spent Nuclear Fuel Loaded Sea Transportation Test Under Normal Conditions of 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 is planned in September 30, 2021. In order to investigate amplification or attenuation characteristics, according to the load transfer path, a number of accelerometers will be attached on a ship cargo hold, cradle, cask, canister, disk assembly, basket, and surrogate fuel assemblies and to investigate the durability of spent nuclear fuel rods, thirty-two strain gages will be attached on surrogate fuel assemblies. The strain gage attachment locations will be same with those of the road transportation tests. A ship named “JW STELLA” which has similar deadweight (5,000 ton) of existing spent nuclear fuel transportation ships is scheduled for the sea transportation tests. The ship is propelled by 1,825 hp two main engines with two 4-bladed propellers. To investigate the vibration characteristics according to the loading position in the deck, three IMU (Inertial Measurement Unit) sensors and three IEPE will be attached to the different locations on deck. In engine room, rpm sensor and one accelerometer will be installed to identify the vibration from the engine and propellers. Those data will be used for the analysis of the sea transportation tests. The effect of ship speed, trim angle, draft will be investigated by changing the operating condition. Even though the weather condition is the most important test parameter, it cannot be controlled. Therefore, the weather data from Korea Meteorological Administration will be used for analyzing the test results. The sea transportation test results will be compared with those of the Multi-Modal Transportation Test.

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

Refurbishment of Non Destructive Testing Bench for Spent Fuel Rod

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The Post Irradiation Examination Facility (PIEF) in KAERI has been operating the non-destructive testing (NDT) bench to perform various examinations, such as visual observations, dimensional measurements, eddy current tests, and gamma scanning, on the spent fuel rods. Since its initial installation in hot cell, as the NDT bench has been utilized for nearly thirty years, the refurbishment of its major components was required to improve the accuracy of test data and maintenance efficiency by introducing the latest equipment. In this research, the technical requirements for the NDT bench were investigated and a number of components were newly developed and installed. The NDT bench comprises the examination bench and the examination station, which has been designed to accommodate full-length fuel rods of various diameters. The examination bench primarily includes the mainframe, carriage module, rotating grip module, and bottom centering-piece module and is driven by several motors for gripping the upper end of the fuel rod and moving the fuel rod vertically. In this refurbishment task, all the older modules were replaced and rearranged, except for the mainframe fixed to the rear door. All the existing motors, DC and stepping motors, were replaced with new stepping motors, and mechanical switches were fixed on each motor shaft to provide movement information. In addition, to ensure that the fuel rod is accurately positioned and to provide data on the fuel rod length, a contact switch was installed in the new centering-piece module. The examination station comprises several test equipment, which is stationary and carries out tests while the fuel rod moves vertically. The measuring instrument equipped with a linear variable differential transformer for measuring the diameter and deflection of the fuel rod and an eddy current array system for detecting any cladding defects and measuring the oxide thickness profile of the cladding were replaced with the latest equipment. The new eddy current array probe comprises multiple coils that enable it to inspect flaws oriented in any direction simultaneously in a single scan. The new control unit and data acquisition system were also installed at the operating area in consideration of the connection with the equipment.

Keywords: Post irradiation examination facility, Non-destructive testing bench, Refurbishment, Spent fuel rod

Estimation of Rigid Body Motion Using Digital Image Correction Method in Drop Test of Transport Container for Radioactive Materials

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A drop test of a transport container for radioactive materials should be performed in accordance with the regulations, and the transport container should be dropped onto a rigid target so as to incur maximum damage. In most of container-type transport containers with a large aspect ratio, the oblique drop causes the maximum damage because a strong second impact occurs due to rotational acceleration after the first impact. Because the impact velocity of the second impact strongly depends on the friction coefficient between the transport container and the rigid target, it is necessary to use the appropriate friction coefficient for an accurate drop analysis of the transport container. In this study, a digital image correction (DIC) method using high-speed cameras is utilized to estimate the rigid body motion of the transport container in the drop test. The impact velocities of the first and second impact and the rotational velocity after the first impact can be obtained using DIC method, and the obtained rigid body motion can be used to estimate the friction coefficient between the transport container and the rigid target for the proper verification between the test and the analysis.

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Keywords: Transport container, Drop test, Drop analysis, Digital image Correction method

Dissolution of Rare Earth Oxide From Uranium Oxide Fuel in the Molten Salt

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As part of a study to develop a method to reduce burden of storage and disposal of high-level radioactive waste, we studied the chlorination process of TRU and RE elements in uranium oxide as a method to remove TRU and RE elements from spent nuclear fuel. Various studies have been conducted on the chlorination of spent nuclear fuel, but UCl_3 , NH_4Cl , and ZrCl_4 were selected as chlorination agent in this research. According to the results of the thermodynamic calculation, there was no difference in Gibbs free energy between the TRU element and the RE element when using each chlorinating agent. Therefore, in this study, the RE element was used as the surrogate of the TRU element.

The simulated fuel containing CeO_2 and Gd_2O_3 was fabricated through molding and sintering process and the powder form was mechanically pulverized to prepare. The dissolution behaviors of rare earth element of simulated oxide spent nuclear fuel in a LiCl-KCl molten salt at 500°C were investigated. Molten salt samples were collected for chemical analysis before and after each reaction, and oxide powder was separated from salt and XRD and SEM measurements were performed. Nearly 50wt% of RE elements in the uranium oxide matrix were dissolved into molten salt by the reaction with each chlorinating agent. This results suggest that the chlorination technique can be used to separate TRU and RE element from the nuclear spent fuels.

Keywords: Chlorination, Spent nuclear fuel, Molten salt, Rare eart

Design of Equipment for Fuel Material Recovery by Oxidative Decladding

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The investigation of fuel material recovery from used nuclear fuel rods (UNFR) has been performed by mechanical decladding methods such as slitting and vibrating techniques. The slitting and vibratory decladding techniques were verified using low/high and low burnup UNFRs, respectively. In the high burnup UNFRs, the slitting decladding efficiency with an increase in the burnup from 52.3 to 54.9 GWd/tU was decreased from 99 to 97.52% due to opening characteristics of the gap between the pellets and cladding. Hence, for recovering higher amounts of fuel material from UNFRs, low temperature oxidation step must be added in the fuel material recovery process.

In the damaged UNFR (DUNFR) with various damaged shapes and burnups, mechanical decladding techniques are difficult to recover fuel material. The oxidative decladding method was evaluated to be suitable for recovering fuel material from DUNFRs. For applying this method, although horizontal rotary kiln-type oxidation furnace has been developed, it was difficult to fill the recovered fuel material into basket and took up a lot of operating space in hot cell.

In this work, for the purpose of oxidative decladding of DUNFRs, we designed the vertical-type oxidation furnace using 3D modelling software. In addition, for preparing the sinterable UO_2 powder, oxidation and reduction treatment of uranium oxides should be repeatedly performed in the same reactor after removal fuel rod-cut basket from reactor. Thus, in order to provide the operation of powder preparation as well as decladding, the main modules such as recovered fuel powder basket, reaction gas distributor, reaction gas flow hood, and fuel rod-cut basket inside reactor were arranged in bottom to top order. The vibrator on the flange was designed to promote both separation of oxidized powder (U_3O_8 or U_4O_9 phase) from fuel rod-cuts and contact between reaction gas and unreacted UO_2 pellets in claddings.

Acknowledgements

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Keywords: Used nuclear fuel, Material recovery, Oxidative decladding, Powder preparation

Performance Evaluation of Forced Gas Dehydration Test System

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As the Spent Nuclear Fuel (SNF) pool in nuclear power plant is expected to be saturated, a method for dry storage of SNF has emerged. A series of work processes to transport SNF from the spent nuclear fuel pool to dry storage facility is called a Short Term Operation Process. This process includes re-loading spent fuel in cask, moving the cask to Decon pit in Spent fuel building, draining the cask, drying and filling with inert gas, and transferring to dry storage. In this paper, the performance of the Forced Gas Dehydration test system manufactured to perform drying during the short term operation process was evaluated. A one assembly scale down canister model is used for the drying performance test. After canister inside is preheated to initial test condition with electric heater and heated circulating gas, the estimated residual water of canister models, 130 g, inject into canister through the drain port. Fill the inside of the canister with helium gas to the test pressure and operate the FGDS, and circulate the helium gas with circulation pump. The water inside the canister is evaporated by the heated circulation gas above 100°C, and helium and water vapor flow together to the FGDS. During the FGDS operation, water vapor is condensed and removed from the system. If the operation set value is reached, FGDS operation stop 30 minutes later. Measure the residual water on the bottom of canister, and calculate the drying rate of FGDS. From the drying performance tests, we have the average drying rate of $3.2 \text{ g} \cdot \text{min}^{-1}$, and expect about 3 hours of drying time for 7,000 liter canister. For the dry storage of SNF, Drying is very important to prevent the deterioration of structures and interior materials in dry storage system, and to secure the storage system integrity.

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Keywords: Forced gas dehydration, Spent nuclear fuel, Dry storage, Canister, Residual water, Short term operation

Oxide Block Fabrication Using CIP

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Spent nuclear fuel generated from CANDU (CANada Deuterium Uranium) and PWR (Presurized Water Reactor) power plants is classified as high-level radioactive waste. Safe management and disposal of spent nuclear fuel are required as the saturation of temporary storage facilities in the pond arrives. In the case of deep disposal of spent nuclear fuel, it is possible to improve disposal safety by separating and collecting major nuclear species that directly affect the disposal area, such as high heat dissipation, high mobility, and long-term radionuclides contained in spent nuclear fuel.

In this study, compactibility experiments using CIP were conducted to manufacture large-scale blocks (> diameter 50 mm) for the disposal of spent nuclear fuel from which high heat dissipation, high mobility, and long-term radionuclides were removed.

CeO₂ powder was used for the experiment as a surrogate of spent fuel powder. For improving compactibility, 2wt% Acrawax was also used as a binder. Mixed powder with CeO₂ and Acrawax was granulated for improving flowability of mixed powder. In using a single press, a green body was not made due to damage to the compacted body, but in using CIP, the green body with the green density of 75% or more of the theoretical density was made. Based on the results of this study, it is possible to manufacture the large-scale oxide block of spent nuclear fuel.

Acknowledgements

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Keywords: Large scale oxide block, CIP, Compactibility, Spent fuel

Analysis of Maritime Accident Cases for Risk Assessment of Domestic Radioactive Waste Transportation

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Currently, the radioactive wastes including low and intermediate level radioactive waste (LILW) and spent nuclear fuel is continuously generated in Korea, and the temporary storage facilities for spent nuclear fuel become saturated, the demand for transportation is expected to increase. Due to this, the importance of transportation risk assessment has increased, and accordingly research on the overland/maritime transportation of radioactive waste has been actively conducted.

Thorough various technologies are required to safely transport the radioactive waste and the radioactive waste transportation risk assessment is one of the important activities related to safety. For the risk assessment, analysis for the case overland/maritime transport accidents should be preceded. Among these, maritime transportation is selected for parameter of this study because bulk transportation for radioactive materials and a low risk of radioactive waste transportation are possible.

In this study, the statistical data of Korean Maritime Safety Tribunal (KMST) was used to review various maritime accidents and the effect of each accident on the radioactive waste transportation risk assessment was analyzed. As a results of analyzing the effect of each accident, both collision accidents and the South Sea, where geographical requirements are complicated had the greatest effect. In addition, the accident rate is high in the shore region near Gyeong-ju, where radioactive waste disposal facilities are located. Those analysis data will be used for the risk assessment data of the radioactive waste transportation.

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Keywords: Radioactive waste, Transportation, Risk assessment, Maritime accidents

Preliminary Verification of Fuel Constituent Redistribution Model

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One characteristic phenomenon of U-Pu-Zr or U-Zr metallic fast reactor fuel during irradiation is the migration of fuel constituents which turns the uniform as-fabricated alloy into a locally heterogeneous fuel. In other words, irradiated metallic fuel shows three concentric microstructural zones containing different U and Zr concentrations in each zone, namely, a porous Zr-enriched central zone, a Zr-depleted and U-enriched intermediate zone, and a slightly Zr-enriched on the outer periphery. This redistribution phenomenon is more pronounced in the case of a ternary fuel (U-Pu-Zr), and a considerable amount of redistribution is also observed in a binary fuel (U-Zr) when irradiated at a high temperature. The redistribution of Zr, a solute element in the fuel slug, occurs not only by the temperature and concentration gradient, but also by the movement of Zr through the phase boundary since there is a multiphase region depending on the history of the fuel slug temperature. This fuel constituent redistribution affects the fuel slug material properties such as solidus and liquidus temperatures, thermal conductivity, mechanical properties such as modulus of elasticity, fuel cladding eutectic reaction rate, and swelling behavior. Therefore, it is essential to perform the fuel performance analysis considering the fuel constituent redistribution.

In this study, the fuel constituent redistribution model was developed using a Zr inter-diffusion flux equation based on Hofman's theory and a simplified pseudo-binary phase diagram for U-Pu-Zr alloys on the assumption of a fixed Pu content along the radial direction. This model was inserted into the LIFE-METAL fuel rod performance analysis code, and the verification was performed by comparing the fuel constituent redistribution measurement results and the code prediction values for the T179 and DP16 fuel rod (U-19Pu-10Zr) in the EBR-II X419 and X441A test, respectively. As a result, it was confirmed that the code prediction values for the Zr concentration distribution along the radial direction of fuel slug match well with the measurement data.

Acknowledgements

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Keywords: Fuel constituent redistribution, U-Pu-Zr fuel, Zr inter-diffusion flux, Pseudo-Binary phase diagram

Interlock Design of Refueling Machine at Nuclear Power Plant

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This study provides interlock design for the Combustion Engineering (CE) type Refueling Machine (RM) and the Westinghouse (WH) type RM at nuclear power plants operating in Korea. The design of the CE RM differs from that of the WH. The fuel handling equipment has a specific handling guideline and its guideline was established based on design of the fuel assembly. In this paper, interlock design of two different type Refueling Machines is reviewed and fuel loading performance for the CE RM and the WH RM is evaluated. To evaluate fuel loading performance for the RM, the first step of fuel loading into the reactor for the RM one cycle starts at the top of the FTS upender with the RM hoist up-limit position. Next step is to grapple a fuel assembly and to raise it at full up to move to the reactor. The final step finishes at the top of the FTS upender for grappling next fuel after the previous fuel loading into the reactor. The total portion of the steps during fuel loading and withdrawal at reactor is an important item for evaluation of loading performance. The difference of the total time between the CE RM and the WH RM depends on the interlock system for the RM hoist and the interlock system between the RM bridge and the FTS upender. As results from the study, the WH type RM is designed to save fuel handling time at reactor by using different interlocks with that of the CE type RM. *Technical information provided in this paper would be* beneficial suggestion to enhance the safety and the efficiency of the fuel handling equipment.

Keywords: Interlock, Refueling machine, Fuel handling equipment, Fuel handling, Refueling, Fuel loading

Multi-Body Dynamic Modeling for Trailer Under Normal Road Transport Condition

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Spent fuel which is generated after the fission process in nuclear power plant must be safely treated due to its risks and importance. In these circumstances, the structural integrity of the spent nuclear fuel under the normal road transport condition was evaluated by performing the test using a commercial trailer with the KORAD-21 transportation cask and dummy spent nuclear fuel. In this research, in order to analytically simulate the vibration and impact features that occur during the normal transport conditions, a multi-body dynamic trailer model was developed based on the test data which is obtained by the normal transport test in 2020. Specifically, the Goldhofer's 10-axle trailer model was used, and the suspension system was assumed to be passive suspension because it is desirable in normal road conditions (lower than $30 \text{ km} \cdot \text{h}^{-1}$). Among a number of scenarios, a total of three transport scenarios (pass over bumps, change lanes, and avoid obstacles) were considered. In the multi-body dynamic trailer model, the driving speed of the vehicle which is obtained by the GPS data was applied as an input, and the usefulness of the trailer model was verified through the comparison with the test results. Finally, a multi-body dynamic trailer model that can follow the test results well was derived.

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Keywords: Multi-body dynamic, Trailer Modeling, Spent fuel transport system, Normal road transportation condition

Experimental Investigation to Prevent Ice Formation in Vacuum Drying

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Spent nuclear fuel generated at nuclear power plants must be safely stored during interim storage periods. To safely store spent nuclear fuel in a dry storage system, residual water must be removed from the dry storage system canisters. The drying methods currently used to remove residual water from dry storage system canisters are vacuum drying and forced gas (helium) dehydration. It is known that ice may form when residual water in a canister is removed using vacuum drying. To confirm a method for preventing ice formation, therefore, a vacuum drying system and a small canister were designed and fabricated. Drying tests were performed in two methods. First, the vacuum drying test was conducted using a vacuum pump with a flow rate of 600 liter/min slower than that of 1,800 liter/min. Then, the vacuum drying tests were conducted by reducing the pressure step by step. As a result of the tests, drying was more done when the hold point was kept at a higher pressure as well as hold time was kept longer. Therefore, it was found that the hold time at each step and the hold point of pressure should be well set to prevent ice formation. In addition, less ice was formed when drying was carried out using a vacuum pump with a slower flow rate. However, the ice formation could not be prevented. Therefore, it is considered that a vacuum drying method that reduces the pressure step by step is more desirable than a vacuum drying method using a vacuum pump having a slower flow rate.

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Keywords: Spent nuclear fuel, Dry storage cask, Vacuum drying, Residual water, Ice formation

A Study on Concrete Constitutive Models for Dry Storage System of Spent Nuclear Fuel

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Concrete has an important role in dry storage systems, such as shielding and physical protection. Damage of concrete structures in the storage systems can have catastrophic consequences. Therefore, a study that can quantitatively evaluate the structural damage is needed. This paper investigates the behaviors of the concrete constitutive models considering the storage environmental conditions of spent nuclear fuel. The strain rate varies depending on the structure's environmental condition, which significantly influences the behavior of the concrete. As the strain rate increases, the concrete strength also increases, and this characteristic should be considered when performing finite element analysis. This paper investigates the respective material models and compares the simulation results using commercial finite element software, ABAQUS, and LS-DYNA.

Among various material models, we considered the concrete damaged plasticity (CDP) model in the ABAQUS, and the continuous surface cap (CSC), Karagozian and Case concrete (KCC) model in the LS-DYNA, representing concrete behavior considering the strain rate effects. This paper shows the specimen scale simulations (e.g. uniaxial compression) to evaluate the effects of parameters of the constitutive models. Based on this study, we have a plan for real-sized structural analysis according to the environmental conditions of the dry storage system.

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Keywords: Concrete, Constitutive models, Dry storage system, Strain rate

Preliminary Evaluation of Cr-coated ATF Spent Fuel Integrity for Dry Storage and Transportation

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ATF (Accident Tolerant Fuel) designed for improving nuclear fuel safety against severe accident condition has been developed internationally after Fukushima disaster. Fuel development could be simply divided into fuel pellet and cladding. The latter is also simply categorized into thin Cr (Chrome) or CrAl (Chrome Aluminium) coated Zircaloy cladding in the near future and a new material-based cladding like FeCrAl or SiC in the long term future. Commercial application of these new concept ATFs need a preliminary safety evaluation for ATF spent fuel because the current spent fuel management trend of dry storage seems to be similarly applicable to the ATF spent fuel management strategy in less than 10 years.

Most important concern of spent fuel dry storage is to keep the spent fuel cladding integrity safely. This is highly related to mechanical integrity when spent fuel is dynamically transported after several decades static dry storage. Hydride reorientation is the most concerned cladding degradation mechanism of mechanical integrity problems because zirconium-based cladding has high oxidation rate and high hydrogen concentration which causes significant radial hydride formation and brittle behavior of cladding.

This study focused on hydrogen behavior of the newly developing Cr-coated ATF cladding and preliminary evaluation of its impact on spent fuel integrity by comparing it with the already existing Zr-alloy claddings.

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Keywords: ATF, Cr-coated, Dry storage, Spent fuel integrity

Fundamental Study on Uranium Oxide Separation From the Spent Nuclear Fuel by Chlorination Using NH_4Cl

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Spent nuclear fuel generated in LWR (light water reactor) and CANDU (Canada Deuterium Uranium) are classified as high-level radioactive waste, and temporary storage facilities on sites are currently being saturated. Therefore, it is necessary to secure key strategic technologies for the safe management and disposal of spent nuclear fuel. The high heat dissipation, high mobility, and long half-life radionuclides contained in the spent nuclear fuel have a direct effect on the disposal area. Consequently, studies to isolate/collect the high heat dissipation and mobility nuclides from actual oxide-spent nuclear fuel or simulated nuclear fuel have been actively conducted. Once these elements are removed from the nuclear fuel waste, the long-lived nuclides such as TRU become an important factor in determining disposal safety and storage area. Here, the TRU along with uranium can be treated and disposed as high-level waste. However, it needs to be noted that uranium accounts for most of the spent nuclear fuel, thus significant volume reduction will be possible by removing uranium. This work focused on separating uranium from the spent nuclear fuel. As a fundamental study, simulated fuel composed of U and RE was used. The simulated fuel was chlorinated by using NH_4Cl , and U was separated from RE using the difference of the chlorination properties. The detailed results will be posted and discussed.

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

Uranium Metal Chlorination Using NH_4Cl for the Preparation of MSR Fuel

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Molten salt reactor (MSR) uses molten-salt-based fuels instead of using solid fuels, which makes the MSR an attractive nuclear reactor with no risk of serious accidents and no requirement of solid fuel fabrication. First design of MSR was proposed at Oak Ridge National Laboratory (ORNL) and two test reactors were manufactured and successfully operated. More recently, interests in MSRs using a fast spectrum (FS-MSR) has been increased and several FS-MSR designs have been under development. For the FS-MSR, chloride-based fuels are considered because chloride salts typically have higher solubility of actinides and lanthanides over fluoride-based salts.

MSR technologies including design concepts, core physics, fuel properties, and corrosion behaviors in the reactors have been heavily studied; however, the manufacture of the molten fuel has not been fully understood. The most feasible method to manufacture the MSR fuel is using the products from pyroprocessing. The products are in the form of metal; therefore, the metals need to be chlorinated. Typically, Cl_2 gas can be used for the chlorination of metals, but it is difficult to treat Cl_2 gas due to its toxicity and U metal can be chlorinated into UCl_5 and UCl_6 form, which is difficult to treat. Another method for the metal chlorination is using chlorination agents such as CdCl_2 , BiCl_3 , and ZnCl_2 dissolved in the molten salt media, which can be react with metals. In this case, it is challenging to produce high concentration of fuel in the salt due to solubility limits, and by-products are additionally generated in the salts.

Therefore, in this work, NH_4Cl was used to produce pure UCl_3 . An airtight reactor was designed to trap NH_4Cl gas phase, which can react U metal inside the reactor. When the chlorination experiments complete chlorinated products were collected and analyzed by XRD and ICP-OES. The detailed results will be further discussed.

Keywords: Molten salt reactor, Molten fuel fabrication, Chlorination, Uranium chloride

Coating Effect of Nuclear Fuel Cladding on Mechanical Integrity of Spent Fuel : Accident Tolerant Fuel

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In many countries, an Accident Tolerant Fuel (ATF) is developing to increase the fuel safety during accident condition and to reduce the fuel failure rate during normal operation. Although there are various concepts in ATF, the most promising near-term cladding technology is coated cladding concept. KEPCO NF and KAERI suggest coating cladding technology to increase high temperature oxidation resistance and mechanical integrity based on current Zirconium alloy cladding.

In addition to high temperature oxidation resistance of Cr (KEPCO NF) and CrAl (KAERI) coated cladding, a very low corrosion layer was observed during normal oxidation condition. As a result of low corrosion rate of coated cladding, it is expected that an ATF cladding can maintain its ductility until discharge from reactor and during storage periods. Since a ductility degradation of cladding is closely related with absorbed hydrogen concentration in cladding during corrosion process.

In recent days, a new project have been started to investigate a ATF fuel with a regulatory point of view, and, an ATF's spent fuel behavior during drying storage is one of the most important topic which must be considered for ATF licensing. In this paper, a coating effect on cladding's corrosion and hydrogen uptake were reviewed and there effect on spent fuel's mechanical integrity were discussed qualitatively.

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Keywords: Accident tolerant fuel, Coated cladding, Corrosion, Hydrogen uptake

Fatigue Damage of Spent Fuel Subjected to Vibration Load Under Land Transportation Condition

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Spent fuel (SF) shall withstand the shocks and vibrations experienced during normal conditions of transportation (NCT) according to 10 CFR 71. In this paper, we evaluated the integrity of SF under land transportation condition by comparing the strain results from finite element (FE) analysis with the O'Donnell-Langer fatigue curve.

The O'Donnell-Langer fatigue curve is universally accepted as the basis for judging cladding integrity under NCT shock and vibration conditions. This curve was expressed in terms of strain versus number of cycles, and used as a strain-based evaluation.

In order to quantify the fatigue damage, the damage fraction was introduced. The damage fraction is defined as the ratio of number of cycles at a given strain amplitude to number of allowable cycles at a given strain amplitude. If the sum of damage fractions reaches 1, failure occurs.

FE model consists of cladding, epoxy and pellets. In order to reflect the SF conditions two things were considered. Firstly, end-of-life (EOL) material properties are applied to cladding and pellets. Secondly, cladding and pellets were bonded with epoxy. Using this FE model and 12-second long acceleration data from land transportation test, axial strains at the outer diameter of the central part of the cladding were obtained.

Strain results from 12-second-FE analysis were categorized according to strain amplitudes using the rainflow cycle counting (RCC) method. RCC was used in the fatigue data analysis to convert various strain spectra to simple combination of strain reversals for analyses of amplitude and cycle easily. 3,000-km travel results were extrapolated from the 12-second results to evaluate the damage of SF under actual land transportation. The cumulative damage with conservative assumptions was 13.5% which is less than 1. Therefore, the integrity of SF will be maintained.

We will carry out a similar procedure to verify the integrity of SF under marine transportation condition.

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Keywords: Spent fuel, Fatigue, Vibration load, Land transportation, Damage fraction

Comparison of the Radioactivity Effects of Mo Additives in the Nuclear Fuel of Light Water Reactor

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The effect of Mo addition pellet and Cr/CrAl coated cladding on nuclear fuel, which is considered an option for accident tolerant fuel, was compared and analysed. The radioactivity value of each nuclide produced as a result of fission was calculated using the SCALE 6.0 Origen-arp code, and the code input value was prepared assuming PLUS7 nuclear fuel model, burnup of 60,000 MWd/tU, cooling time of 6 month and one year.

Mo additive pellet was compared with the conventional commercial fuel pellet to gamma spectra in all categories, including light element, actinide, fission product using the Origen-arp code. In addition, the decay chain of the radioactive nuclides produced by the reaction of additive elements and neutrons was reviewed. As a result, the effect of Mo additives and Cr/CrAl coating on the change in radioactivity was evaluated to be insignificant compared to the effect of reducing the amount of Uranium by the additives.

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Keywords: Accident tolerant fuel, Molybdenum, Light water reactor fuel, Origen-arp

Calculation Procedure of Vacuum Drying for Canister

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The first step in dry storage of spent nuclear fuel is the drying process. A common drying method is vacuum drying. During the drying process, only the pressure change inside the canister could be measured. It is impossible to evaluate how much residual moisture is and how much moisture is evaporated in the canister.

The vacuum drying process was evaluated by an analytical method to evaluate the non-measurable part. Analytic approach to simulate vacuum drying was suggested. Simple vacuum drying process and repeated vacuum drying process were analyzed.

1. In simple drying process, only pressure change from 760 Torr to 3 Torr is focused. Dryness criteria is not considered. The purpose of simple drying process is to check possibility of vacuum drying simulation. Vacuum drying is based on pressure change in canister due to vacuum pump and water evaporation.

To check the effect of residual water, case with residual water and case without water were compared. Required time until pressure was less than 0.4 kPa (3 Torr) is 111 s for result without water and is 35907 s for result with water. It took much longer time for vacuum drying of case with residual water.

2. In repeated drying process, dryness criteria was considered. To meet dryness criteria, vacuum pumps operated until pressure went down under 3 Torr. Even if pressure was less than 3 Torr, pressure could rebound more than 3 Torr because very few residual water could evaporate and it led to increase pressure. This phenomenon is called as pressure rebound. To avoid pressure rebound, drying process should be performed several times. To meet dryness criteria, vacuum drying process was repeated 24 times and it took 26.5 hours. Pressure rebound could be found at each drying stage.

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Keywords: Vacuum drying, Analytic approach, Spent nuclear fuel, Dry storage

Study of Spent Fuel Prototype Source Manufacture for Gamma Emission Tomography

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The nuclear fuel cycle ends with the safeguards, secure and sustainable management of the spent fuel, which includes its storage after withdrawal from the core of the nuclear power plant, followed by either its processing/recycling or final disposal. Safe, secure, proliferation resistant and economically efficient nuclear fuel cycles that minimize waste generation and environmental impacts globally contribute to the sustainability of nuclear energy.

For this reason, technology development is required to continuously manage and monitor spent nuclear fuel, and this must be verified.

However, spent nuclear fuel is classified as high-level radioactive waste according to the domestic regulatory system because of its high concentration of generated heat and radioactivity. In addition, because of its long half-life, it takes hundreds of thousands of years to decrease to the level of natural uranium that can be directly measured based on gamma characteristics. Therefore, it is impossible to verify the measurement device using the spent nuclear fuel directly, and CRM cannot be created either.

In this study, the type and frequency of prototype spent nuclear fuel required to verify the spent nuclear fuel measurement device were investigated and the production requirements for this were presented. The current status and type of domestic storage of prototype sources were studied, and using this, prototype sources with representation were selected to derive a production method based on the physical and chemical characteristics of prototype sources. In addition, legal review and economic evaluation of manufacture were also performed.

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Keywords: Spent fuel, Fission product, Fuel rod, Prototype source, Safeguards

Phosphate Precipitation of Uranium and Rare Earth Chlorides in Molten Salt

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The Pyro-SFR fuel cycle can maximize the utilization of actinide resources and minimize the waste burden. Under the circumstance of energy transition policy of current government in which nuclear energy portion will be gradually reduced, some R&D study looking for alternatives other than Pyro-SFR recycling could be very valuable and timely suitable. In this regards, new alternative study started to evaluate the possibility of it if there are some advantages in terms of waste burden in case that the spent fuel is appropriately treated and disposed of in a disposal site, instead of recycling of spent nuclear fuels (SNF). The alternative study separates the fission products (minor actinides and rare earths) from SNF in a molten salt media.

The molten salt coming from the alternative study is radioactive and heat-generated because it contains the fission products chlorides. The waste molten salt is not compatible with a final waste form for permanent disposal. Therefore, it is necessary to collect the fission products from the waste molten salt for minimization of the high-level waste volume and to generate a final waste form containing the fission products compatible to the disposal site.

In this study, based on the results of a review for various precipitation methods such as oxidation, phosphorylation (phosphate precipitation), reduction of metal ion in a molten salt, phosphorylation method of metal chlorides selected as a proper precipitation treatment of the fission products in SNF. Phosphate precipitation has the potential for removing molt of fission product elements from a molten salt arising from the treatment of spent nuclear fuel. A previous KAERI's study on phosphate precipitation of the rare earth chlorides in waste salt shows very good performances from the viewpoint of the conversion ratio (0.99) and precipitation rate (within 10 min). In order to evaluate the phosphate precipitation behaviors of multi-component salt mixed with the minor actinides, a molten salt containing uranium as a surrogate of the minor actinide elements and three dominant rare earth (Nd, Ce, La) chlorides were tested using a phosphate precipitation method at an experimental condition (500°C, 100 rpm, 1 eq. of phosphorylation agent). A cyclic voltammetry (CV) diagnostic method could monitor in-situ phosphate precipitation behaviors for determining the precipitation rate and conversion ratio be evaluated using ICP analysis results of the salt samples.

Keywords: Uranium, Rare earths, Precipitation, Phosphate, Spent nuclear fuel

Preliminary Research for Developing a Reliable Spent Fuel Analysis Model

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The safety analysis report submitted for design of transport casks mainly evaluates transport accident conditions. In the case of structural evaluation, validations of containment and shielding performance of transport casks are performed under transport accident conditions such as drop, puncture, fire, and water immersion. Under the transport accident condition, the integrity evaluation of the spent nuclear fuel is not performed because it is assumed that all of the spent fuel is damaged. On the other hand, as the saturation of wet storage pool capacity of domestic nuclear power plants is imminent, transporting the spent nuclear fuel to the dry storage facilities is becoming a reality in the near future. Therefore, in addition to maintaining the integrity of the transport cask itself, we are carefully studying whether the spent nuclear fuel will remain integrity during the normal transport process and storage condition.

Therefore, it is necessary to develop a reliable analytical model for spent nuclear fuel. In order to develop a reliable analysis model, The effect of degradation factors on the structural integrity of spent nuclear fuel was evaluated. That is, radiation embrittlement, fatigue, oxidation, pitting corrosion, galvanic corrosion, stress corrosion cracking, which are aging mechanisms affecting zirconium-based spent fuel cladding. The phenomena of hydride reorientation, delayed hydride cracking, thermal and low-temperature creep, and local mechanical overload were analyzed.

As a previous study, the development procedures of the analysis model of the spent fuel assembly conducted by EPRI, SNL, and NRC were analyzed. In order to evaluate the damage of the spent nuclear fuel loaded in the transport cask, it is necessary to perform an evaluation of the load path from the transport cask to the spent fuel. Using an analysis model that includes all structures at once is not practical in many cases because it increases the computational time and includes many structural discontinuities. Therefore, in order to solve this problem, it seems desirable to set the direction of the analysis model development by introducing a sequential analysis technique or developing a simplified model of the spent nuclear fuel from sophisticated structure model.

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Keywords: Reliable analysis model, Spent nuclear fuel, Degradation, Storage

A Generalized Block Model for Processing Options of the Nuclide Management Process

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To find alternatives reducing the environmental burden on direct disposal of spent fuels, KAERI is conducting an option study on the nuclear fuel cycle (NFC) system linking the nuclide management process and disposal. The core of the option study is to select and quantify evaluation criteria for evaluating NFC alternatives. Past option studies have evaluated NFC alternatives based on quantitative evaluation criteria such as effectiveness (disposal volume, disposal area), safety (exposure dose), and economics (cost), even if they contain many assumptions and uncertainties. To this end, it is necessary to calculate the mass flows and waste properties for NFC stages, and to convert them into quantitative evaluation criteria through engineering approaches.

Since the purpose of developing the nuclide management process is to increase the connectivity and efficiency with disposal technologies, the processing stages, options and performance goals are not fixed as one. This means that since various processing options exist, multiple process flows appear, and the mass flows and waste properties for each process flow must be re-modelled and calculated. This can be a new and challenging part compared to previous option studies in which the mass flows were derived from one process flow and performance goal. In this project, this part will be overcome through generalized block modeling, which is the generalization of the mass flow and waste property calculation models for the numerous processing options that may exist in the process into one model. Therefore, one process flow can be easily modeled by simply attaching the generalized processing option block and entering different parameters for each option. This may be more applicable in superstructure modeling for simultaneously consider several process flows in the future.

In this study, a draft of a generalized block model for the mass flow and waste property of the process option is proposed. A block model is being built in the GoldSim interface to facilitate the NFC system analysis. Using the developed block model, a simple example of the nuclide management process is simulated to verify its validity.

Keywords: Nuclide management process, Mass flow model, Waste property model, Generalized block model, GoldSim

Development of Continuous Casting System for Annular Metallic Fuel Fabrication

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Innovative metal fuel manufacturing original technology development task is a task to develop core base technology for metal fuel manufacturing technology with high safety and combustion. High performance metal fuel manufacturing process that can reduce process waste by omitting sodium filling process and simplifying the casting process. It aims to secure innovative technologies for metal fuels by developing them.

Continuous casting is one of metallic fuel fabrication technologies, which are applicable for next generation fuel such as small modular reactor (SMR). Various types of metallic fuels such as rods, plates, and annular can be produced by the continuous casting process. This casting process has advantages of reducing a loss of nuclear materials and diminishing the production of radioactive wastes during casting. KAERI is developing a continuous casting technology of an annular type fuel to apply the next generation of metallic fuel.

Based on the results of the previous continuous casting test, the drawing speed and cooling jacket mold were designed, and the continuous casting device driving verification test was performed. The preliminary manufacturing test was succeeded by securing the casting conditions such as the drawing-out casting speed, the melt temperature and the cooling rate, and the process parameters that control the flow of the melt by using the vertical continuous casting equipment possessed.

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Keywords: Continuous casting, Annular fuel, Metallic fuel, Casting process

3분과

고준위폐기물처분 (Oral)



Achieving Local Mass Conservation When Using Continuous Galerkin Finite Element Methods to Solve Solute Transport Equations With Spatially Variable Coefficients in a Transient State

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Mass balance checks are often used to evaluate the abilities of numerical simulators and assess the accuracy of the solutions but are often insufficient. In addition, there is a need to calculate the difference in mass fluxes, through downstream subdomain analysis, to determine whether remediation technologies are suitable for upstream contaminated sites. The Galerkin finite element method (GFEM) has been commonly considered locally nonconservative, whereas the finite difference method (FDM) is a local mass conservation method. In this study, we proved that the GFEM has local mass balance equations on “mass conserved elements” (constructed by connecting lines that perpendicularly cross element faces at midpoints) in solute transport problems with spatially variable velocities and dispersion coefficients, identical to discretized balance equations in the FDM. Furthermore, we found that locally conservative GFEMs with two types of approaches involving spatially variable coefficients (i.e., a linear combination approach and an average approach) give rise to two types of postprocessing techniques for calculating local mass fluxes. These are equivalent to an area-weighted method and first-order Taylor series method, respectively. Finally, to illustrate the practical applicability of these GFEMs, we used the proposed postprocessing approaches to resolve practical problems involving computation of local mass fluxes, based on information directly obtainable from finite element solutions of concentration distribution. Our findings revealed that the local mass balance errors were negligibly small, regardless of numerical concentration accuracy.

Acknowledgements

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Keywords: Galerkin finite element method, Local mass conservation method, Solute transport problems, Linear combination approach, Average approach

Estimation of Evapotranspiration Using Temperature-based and Radiation-based Methods

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The geological disposal of high-level radioactive waste is one of the important problems to be solved worldwide, and for the safety of the geological disposal, evaluation of long-term changes in the geological environment is essential. Changes in the surface environment, such as climate change, can affect hydrological factors such as evapotranspiration or runoff. In particular, it is reported that 70% of rainfall is lost due to evapotranspiration. This means that changes in the amount of evapotranspiration due to climate change can affect the hydrogeology of the geological formation where the disposal facility is located by changing the amount of water to be recharged. Therefore, a detailed evaluation of evapotranspiration is required. Several previous studies have been conducted on the method for calculating evapotranspiration using meteorological factors, and the FAO (No.56) selected the Penman-Monteith method as the standard equation. The Korea Meteorological Administration has also been using the Penman-Monteith method to calculate evapotranspiration since July 2016. However, many meteorological factors are used in the Penman-Monteith equation, and it is difficult to derive evapotranspiration even if only one of these factors is missing. Therefore, this study was conducted to identify appropriate methods for estimating evapotranspiration in each region in Korea when meteorological data are limited.

Meteorological data (temperature, wind speed, humidity, solar radiation, vapor pressure, sunshine ratio, daytime) for 10 years from January 1, 2011, to December 31, 2020, were used. Evapotranspiration was estimated using temperature-based methods (Blaney-Criddle method, modified Blaney-Criddle method, Hargreaves-Samani method) and radiation-based methods (Simple Abtew method, Makkink method, Priestley-Taylor method, Turc method, Solar radiation-Maximum temperature method). The values calculated using these simple methods and the values calculated through the Penman-Monteith equation provided by the Korea Meteorological Administration were compared, and the methods that were most similar to Penman-Monteith calculations using the slope and correlation coefficient were selected for each region. This will be important data for deriving reliable evaluation results by minimizing the omission of evapotranspiration-related data in the surface water-groundwater interface model in the area that will be the site of high-level radioactive waste disposal facility of Korea in the future.

Keywords: Evapotranspiration, Evapotranspiration estimation, Temperature-based methods, Radiation-based methods

Development of the Three-dimensional Discontinuum-based Coupled Hydro-mechanical Simulator for Modelling a Deep Geological Repository

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Crystalline rock is considered as a host rock for the deep geological repository for high-level radioactive waste due to its low permeability, high mechanical strength and high chemical stability. However, the crystalline rock can contain the discontinuities depending on the brittleness, and the discontinuities can accompany the mechanical instability and additional flow path. Therefore, the hydro-mechanical behaviors of discontinuities should be characterized before modelling the crystalline rock based disposing systems. Discontinuum model can generate numerous discontinuities and reflect the detailed behaviors of fractured rock, so it is often used to simulate the fractured and jointed rock masses. However, the coupled hydro-mechanical simulation modules of the currently developed discontinuum codes have several limitations: simplified hydraulic/mechanical analysis, limited applicability, exclusively for two-dimensional analysis and low calculation efficiency. In this research, the discontinuum-based coupled hydro-mechanical modules are developed by linking two different codes to overcome the previous limitations. The discontinuum models are generated by discrete element methods which handle the mechanical simulations and converted to the finite difference models for hydraulic simulations. The discontinuities of discrete element models are assumed as narrow elements with aperture-size width in the finite difference model. During the iterations, the porosity and permeability of discontinuities are updated according to the deformations. Developed simulators can describe the hydro-mechanical behaviors of blocks and discontinuities and be expanded to the thermal and chemical analysis by linking additional simulation modules. The discontinuum-based simulators can construct the entire multiple barrier systems of a deep geological repository and simulate the interaction between each barrier system by the coupled processes.

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Keywords: Discontinuum model, Numerical simulation, Coupled processes, Geological repository

Chemical Transport-controlled Model for Estimating Cu Corrosion Depth

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Cu canister is a primary barrier to prevent the release of radionuclides from used nuclear fuel to the environment. The integrity of this barrier needs to be maintained for long periods of time. Since experiments cannot be performed for such long periods, it is necessary to develop a computational code to predict the corrosion behaviors of this canister. The physical changes and chemical reactions of these complex chemical systems in a deep geological repository must be reflected to the code. As the first step, we develop a transport-governing method to simulate uniform corrosion based on finite element methods. The transport-governing model estimates corrosion depth through oxidants (O_2 , HS^-) flux on the copper surface. The initial concentration of O_2 are same value at atmospheric conditions, which is a conservative assumption. The concentration of HS^- on the boundary of host rock and engineered barrier is 3 ppm. In addition, for accurate estimation of corrosion depth, mutual relationships of thermal-hydraulic behavior are coupled with the chemicals transport model. In order words, the effects of temperature and water saturation on the diffusion coefficient of oxidants are applied.

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Keywords: Copper corrosion, Thermo-Hydro-Chemical model, Transport-governing, Deep geological repository

Evaluation of k_{eff} and End Effect of CE 16×16 Spent Fuel Assemblies on MPC-32 for Long-Term Period Cooling

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Safe management of spent nuclear fuels (SNF) is one of the major issues in South Korea. In the transportation, storage and disposal of spent nuclear fuel, sub-criticality must be maintained, and burnup credit is considered in criticality safety analysis for efficient transportation and storage. In this study, k_{eff} and end effect, which vary depending on the cooling period, were evaluated for CE 16×16 type fuels for MPC-32 casks, which are multi-purpose canisters during transportation and storage applying for burnup credit. In particular, the inventories of major nuclides and their effects of the nuclides on reactivity were evaluated.

The SCALE 6.1/STARBUCS code was used for the depletion and criticality evaluation. The evaluation was conducted by considering 10,000 active cycles, 100 inactive cycles, and 2,000 particles. The CE 16×16 spent fuel assemblies discharged from Hanbit Unit 3 were considered. They had an average U-235 enrichment of 4wt%. The burnups of the SNFs were divided into three groups: 1) Group 1: 50–54 MWd/kg, 2) Group 2: 38–42 MWd/kg, and 3) Group 3: 18–22 MWd/kg. For each burnup group, the analysis was performed with bounding axial profile giving highest reactivity. The bounding axial burnup distribution used for end effect evaluation was determined by calculating k_{eff} using various axial burnup profiles of the discharged SNFs evaluated by core following calculation based on the Hanbit Unit 3 nuclear design report. In addition, the k_{eff} comparison of values from criticality analysis with burnup credit were evaluated considering major actinide nuclides set, and actinides plus fission product nuclides set over a long cooling time of 100,000 years.

In the three burnup groups, the reactivity decreased until 100 years, but the reactivity increased from 100 years to 5,000 years, and then the reactivity decreased to 100,000 years. The k_{eff} of all three burnup groups were evaluated to be a similar or lower level than the k_{eff} obtained with a minimum cooling period of 5 years for transportation. As expected, the end effect was the highest in the 50–54 MWd/kg burnup group. The highest end effects were found at the cases 100,000 years cooling but their k_{eff} values was the smallest. The end effects for considering only actinides and actinides plus fission products were evaluated to be 3,000 pcm and 5,000 pcm considering actinides, respectively. The inventories of Am-241 and Gd-155 nuclides increased to about 100 years, and thus the reactivity decreased to about 100 years. Gd-155 has not fluctuated significantly since 100 years, but the reactivity has decreased due to the continuous increase in Am-241 inventory and then increased at 5,000 years due to the decay of Am-241. Then, the reactivity slightly decreased from 5,000 to 20,000 years, and then the reactivity rapidly decreased to 100,000 years due to the decay of Pu-239, which is a fissile material. From the analysis, it was found that the CE 16×16 type had a margin in k_{eff} even when at least 5 years of cooling, which is the transportation regulation. However, such kind of analysis will be required for the other casks and SNFs to check the criticality safety over long cooling period.

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Keywords: Burnup credit, Cooling time, End effect, Spent fuel, Criticality

Development of Through-diffusion Equipment to Estimate Diffusion Coefficients of Radionuclides in Rocky and Buffer Materials

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Currently, most reasonable concept for the high-level radioactive waste (HLW) disposal is to adopt multi-barrier disposal system including engineered and natural barriers in a sufficient depth of geological environments. In a long-term perspective, the groundwater can infiltrate the underground disposal site and promote corrosion of metal canister containing HLW. In this scenario the radionuclides can be released from the canister by the groundwater. Therefore, the bentonite buffer surrounding canister and crystalline bedrock around the geological repository are important to retard a migration of radionuclides. The diffusion process is one of solute transport ways in deep underground environment. In order to evaluate a migration of radionuclides in the safety assessment or geochemical modelling, diffusion coefficients of radionuclides are required as input data. The through-diffusion experiment is commonly adopted to study the diffusive migration of radionuclides in rocks and bentonite. To determine the effective diffusion coefficients of radionuclides in rock matrix and bentonite buffer, we have developed a through-diffusion equipment. The equipment basically consists of two reservoirs between diffusion media such as rock or bentonite disks. One reservoir with a relatively large volume contains high content of radionuclide in a synthetic groundwater and the other reservoir with a smaller volume contains only the synthetic groundwater.

We have designed two kinds of the through-diffusion equipment to obtain diffusion coefficient for crystalline rock (i.e., granite) and bentonite (KJ-II), respectively. The disk-shaped crystalline rock by cutting drilling core and compacted bentonite ($1.6 \text{ g} \cdot \text{cm}^{-3}$) are installed in the equipment. The rock sample can be easily fastened by simple method because of its hardness. In contrast with rock, due to the swelling characteristic of bentonite under wet condition, we have newly designed a diffusion cell using stainless steel that can resist a swelling pressure. The compacted bentonite is placed in the middle of diffusion cell with metal porous filter and perforated plate to prevent loss and corrosion of the bentonite sample. The diffusion experiment would be conducted to evaluate diffusion coefficients of radionuclides from the changes of solute concentration in the reservoirs according to planned time. Those experiments are expected to contribute understanding the radionuclide behaviors in geological disposal environments.

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Keywords: Through-diffusion equipment, Diffusion coefficient, Crystalline rock, Bentonite, Geological disposal environments

U-series Disequilibrium and Stability of Uranium Minerals: Natural Analogue for High-level Radioactive Repository in Granite

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Uranyl carbonate minerals, known for its high solubility in groundwater, were occurred as fracture filling materials (FFMs) in the granitic host rock in KAERI Underground Research Tunnel (KURT) site. In this study, we examined the long-lived radioactive isotopes such as ^{230}Th , ^{232}Th , ^{234}U , ^{235}U and ^{238}U and the rare earth elements (REEs) in the FFMs to understand the behaviors of the uranium. The distribution of $^{235}\text{U}/^{234}\text{U}$ ratio indicated that the uranyl carbonate minerals were deposited from the ancient low temperature groundwater. The vertical distribution of U fractions in fracture filling materials was heterogeneous according to the complex dissolution and precipitation of uranyl carbonate in fractures. The uranium minerals were preferentially deposited in the redox transition area between the deposit dominant zones, of which ages were between 144 Ka and 164 Ka. The depths of the uranium precipitates in fracture filling materials ranged from 50 m to 203 m, bgl. This study demonstrated that the uranyl carbonate minerals were conserved and can be used as a indicator of the shallow depths of redox complex zones for the geological time scales.

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Keywords: Uranium isotopes, Fractured granite, Uranyl carbonate, Fracture filling material, REE, Redox

A Review of the Development of Diffusion Databases for Safety Assessment of Deep Geological Disposal in Various Countries

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A safe management of the high level radioactive waste, which includes the spent nuclear fuel and its contaminated products from the subsequent chemical treatments, has become a serious question at issue in the nuclear industries of many countries. The high level radioactive waste is composed of various radionuclides, such as actinides, technetium, cesium, strontium, technetium, and iodine, and they should be completely contained and isolated from the ecosystem for long-periods of time (e.g., 10^5 to 10^6 years) until the overall radioactivity becomes sufficiently low to the level of natural uranium. In this framework, a concept of deep geological disposal, which is based on the multiple-barrier system consisting of the oxide fuel matrix, canister, buffer, tunnel backfill, auxiliary components, and natural bedrock, has been the most widely considered and pursued design for a permanent disposal of the high level radioactive waste. Furthermore, in order to reliably assess the long-term performance and safety of a deep geological repository, the migration behaviors of radionuclides, such as the solubility, precipitation, sorption, and diffusion, should be explicitly comprehended under the relevant geochemical conditions in each repository component. Among the complicated behaviors, diffusion is a chemical mechanism by which the species progress as a result of the development of chemical gradient (e.g., concentration gradient), and it is the predominant process for transport of dissolved radionuclides and corrosive substances in the buffer, tunnel backfill, and microporous structure of crystalline bedrock since porosities and permeabilities of the barrier matrices are expected to be effectively low. In addition, the diffusion mechanism also plays an important role in the physical and chemical evolutions of repository system, such as the mineral transformation and alteration of pore waters. In this regard, various organizations responsible for the disposal of radioactive waste in each country have developed their own diffusion databases which includes the compilation of effective and apparent diffusivities of radionuclides in the barrier matrices to perform a reliable safety assessment of the deep geological disposal. The present work provides a brief review of the numerous diffusion databases introduced so far and suggests further approaches for the development of radionuclide diffusion database relevant to the domestic environment.

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Keywords: Radionuclide migration, Buffer, Tunnel backfill, Geosphere, Diffusion, Database

3분과

고준위폐기물처분 (Poster)



Absorption of U(VI) Surface Species on SiO₂ Sediments

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Silicon dioxide (SiO₂) is an abundant component of geological media and clay minerals. The adsorption of U(VI) onto SiO₂ is widely studied to understand its transport behavior. The information of distinct U(VI) surface species has been investigated using versatile luminescence spectroscopy. Absorption spectroscopy is also beneficial in the research field of U(VI) sorption, however, it is difficult to apply to the opaque substances, such as several clay minerals and metal oxides. In this study, the absorption properties of U(VI) surface species on opaque SiO₂ sediments were measured using two different techniques: excitation-emission matrix (EEM) spectroscopy and laser-induced photoacoustic spectroscopy (LPAS). EEM spectra were measured using a spectrofluorometer (Horiba Scientific, Aqualog) equipped with a 150 W ozone-free xenon arc lamp. Excitation spectra were recorded by scanning the excitation wavelength from 200 to 600 nm with a 2 nm step. Emission spectra were measured using a CCD in the wavelength range of 242–823 nm with a resolution of 1.16 nm. A wavelength-tunable optical parametric oscillator (Continuum, Horizon) pumped by a pulsed Nd-YAG laser beam at 355 nm (Continuum, Surelite) was used for LPAS. The LPAS signals were generated by using a piezoelectric transducer. A high-purity grade SiO₂ gel with a particle size of 40–63 μm (Sigma Aldrich 227196) was used to prepare SiO₂ sediment with particle concentration of 10 g · L⁻¹. Specific surface area determined by the N₂ BET method was approximately 477 m² · g⁻¹. The adsorption reactions of U (concentrations from 3.4–100 μM) onto the sediment were performed in 50 mL of a mixed solutions at pH 7.5. It was observed that the U(VI) surface species on the wet sediment showed clearly different absorption peak maxima (around 230 and 290 nm) compared with those (around 325 and 425 nm) measured from the dry sediment. These results suggest the existence of different U(VI) surface species between the wet and dry environments. We will discuss the application of these techniques to study the sorption characteristics of U(VI) on bentonites at a molecular scale.

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Keywords: Absorption, U(VI), SiO₂, Sorption, EEM, LPAS

Diffusion and Sorption of Some Typical Sorbing Nuclides Onto Concrete Upon Chemical Degradation

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Diffusion and sorption behavior of THO, Sr, Cs, and Am onto concrete was investigated with a through-diffusion setup to understand transport processes around the repository silo. The concrete and groundwater were sampled from the radioactive waste repository site. The diffusivities of the nuclides were obtained by a linear curve fitting of the diffused concentration curves with time. After one and a half-year diffusion experiment, the concrete coupons were recovered. In addition, a sequential chemical extraction was conducted to estimate the sorption types of the nuclides for the recovered concrete.

The relationship between diffusion and sorption was investigated in the aspect of sorption reversibility. Sr and Cs showed reversible sorption tendency, while Am had a high and strong irreversible sorption with a very low diffusivity. From the results of the sequential chemical extraction after the diffusion test, some hints were obtained regarding the relationship between the diffusion and sorption. When the nuclides pass through the concrete pores, Sr and Cs sorb mainly by the ion exchange reversibly, while Am sorbs mainly on the iron oxides or fix to the mineral structure of aggregates in the concrete. The most distinct result is the strong bond of Sr on the concrete. Substitution of Sr for Ca in the cement hydrates is the main sorption mechanism as well as Sr showed a certain portion of strong and irreversible bond by lattice incorporation in the cement hydrates.

During the diffusion test, owing to the chemical degradation of the concrete, some precipitations and suspended matters were observed and identified as Mg-Si-O compound and CaCO_3 by SEM and EDS analysis. The obtained apparent diffusivities of tritium, Sr, Cs and Am were about 2×10^{-11} , 9×10^{-16} , 4.4×10^{-16} , and $1.5 \times 10^{-18} \text{ m}^2 \cdot \text{s}^{-1}$, respectively. They are a little lower value than those of the other crystalline rocks except Cs. The strong alkalinity in the pores of concrete may prevent diffusion of cations.

Keywords: Diffusion, Sorption, Concrete, Degradation, Sequential extraction, Strontium

Estimation of Disposal Spacing and Rock Mass Conditions for High-efficiency Repository Based on Temperature Criteria of Bentonite Buffer

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This study investigates the maximum temperature and rock mass conditions through coupled thermo-hydro-mechanical numerical simulations for securing mechanical stability of the high-efficiency radioactive waste repository if temperature criteria of bentonite buffer is 100°C and 125°C, respectively. The maximum temperature at the interface between canister and bentonite buffer is calculated to be 99.4°C, in case that temperature criteria of bentonite buffer is 100°C and disposal tunnel spacing and deposition hole spacing are 40 m and 5.5 m, respectively. The maximum temperature at the bentonite buffer is 99.8°C, when tunnel spacing and deposition hole spacing are changed to be 30 m and 6.5 m. If temperature criteria of bentonite buffer is increased to 125°C, spacings of disposal tunnel and deposition hole is reduced to 30 m and 4.5 m, respectively, which decreases the disposal area by 55% in comparison with the disposal area of Improved Korean Reference Disposal System (KRS⁺). Mechanical stability of high-efficiency repository is evaluated for various disposal spacings based on the Mohr-Coulomb failure criterion. In order to prevent failure of rock mass, RMR of rock mass for KRS⁺ should larger than 72.4, which belongs to good rock in RMR classification. Required RMR of rock mass for mechanical stability is increased as disposal spacing is reduced. RMR larger than 87.3 is needed for a case with disposal tunnel spacing of 30 m and deposition hole spacing of 4.5 m. If the enhancement of rock mass strength due to confining stress induced by swelling of the bentonite buffer and backfill is considered, mechanical stability of the repository is secured for all cases with RMR over 75.

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Keywords: High-level radioactive waste disposal, Temperature criteria of bentonite buffer, High-efficiency repository, Coupled thermo-hydro-mechanical behavior

Thermal-Hydro-Mechanical-Damage Coupled Modeling of the Wolseong Low and Intermediate Level Radioactive Waste Disposal Silo

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Structural damage of underground waste disposal structures due to groundwater and decay heat can result in decreased disposal safety and possible radionuclide transport. Radioactive waste disposal structures are subjected to coupled thermal-hydro-mechanical conditions and identifying concrete damage under such conditions is crucial to ensure safe long-term disposal. In this study, a thermal-hydro-mechanical-damage (THMD) coupled model is developed and applied to the Wolseong Low and Intermediate Level Radioactive Waste Disposal Center (WLDC) silo to investigate damage evolution in concrete waste disposal structures. The Mazars' damage model is adopted to quantify compressive and tensile damage in concrete and rock. Cylindrical concrete specimens were manufactured based on the same mix proportion as the concrete used in the WLDC silos. The concrete specimens were initially water cured for 28 days, followed by a further 32 days of dry or wet curing at two different temperatures (15°C and 75°C). The Mazars' damage model parameters were obtained via uniaxial compressive and Brazilian tensile tests after 60 days of curing. The Mazars' damage model parameters for rock were assumed from previous experimental studies on KURT granite. A two-dimensional model of the WLDC silo was modeled using the finite element analysis software COMSOL and the Mazars' damage model was applied to the silo concrete and surrounding rock. The Richards equation and heat transfer in porous media modules were used to simulate hydraulic and thermal analysis respectively. Waste disposal loading was applied at the bottom of the silo structure and the damage evolution characteristics were investigated for different temperature, saturation and loading conditions. The analysis results displayed increased damage propagation at the silo-rock and silo-backfilled construction tunnel interfaces with increased loading and larger initial damage propagation with saturation. The results highlight the importance of structural damage when considering the long-term stability and safety of underground radioactive waste disposal structures under coupled thermal-hydraulic conditions.

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Keywords: THMD coupled analysis, Mazars' damage model, Radioactive waste disposal, COMSOL

Effects of Bentonite Penetration on the Mechanical Properties of KURT Granite Rock Mass

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The KAERI reference disposal system for high-level waste and spent nuclear fuel consists of an engineered barrier system (EBS) constructed in hard rock at a depth of several hundreds of meters below the ground surface. After the emplacement of the spent fuel into the deposition hole, a compacted bentonite buffer is installed around the spent fuel cannister. With groundwater saturation, the compacted bentonite buffer swells and exerts a pressure onto the inner walls of the deposition hole. The swelling pressure causes the saturated bentonite to penetrate into the excavation-induced rock fractures that exist around the inner walls of the deposition hole. The presence of bentonite in the rock fractures significantly affects the physical and mechanical properties of the host rock mass. Wave-based exploration techniques have been widely used to investigate the effects of gouge fill on rock mass properties.

In this study, the effects of bentonite penetration on the mechanical properties of jointed rock mass are examined using the quasi-static resonant column test. Eleven KURT granite discs were stacked to simulate an equivalent continuum jointed rock mass with regularly spaced planar joints and to ensure long-wavelength propagation. Internal and external plastic membranes were installed between each rock disc to secure the Gyeongju bentonite gouge fill. The long-wavelength longitudinal and shear elastic wave velocities of the jointed rock mass and bentonite gouge-filled rock mass were obtained for different axial stress and saturation conditions. The changes in normal and shear joint stiffness with bentonite penetration were deduced based on the equivalent moduli of the equivalent continuum jointed rock mass.

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Keywords: Quasi-static resonant column test, Rock joint, Bentonite buffer, Elastic wave velocity

Theoretical Electrical Resistivity of Blocked Buffer Materials for HLW Repository

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High-Level radioactive waste (HLW) is planned to be disposed in deep geological repository composed of multi barriers including canister, buffer, backfill, plug, gap filling materials, and near-field rock mass. Among those multi barriers, the buffer plays diverse roles such as preventing the groundwater from infiltrating to canister and releasing the decay heat from the canister to near-field rock mass. The buffer encloses the canister by piling up the compacted bentonite blocks. If the buffer is damaged (e.g. cracked, channeled), integrity of the whole repository is affected. Therefore, it is necessary to determine the buffer block fulfill the design criteria with no damage of the buffer. The electrical resistivity is well-known geophysical method that is affected by geometries of electrode and specimen and material properties. Material properties such as porosity and degree of saturation (can be converted into water content) have a strong relationship with the electrical resistivity. The relationship between material properties of buffer material and electrical resistivity is now starting and the methodology to estimate the electrical resistivity of the buffer block is suggested in this study prior to the correlation. Theoretical equations are derived for the buffer block and they are verified with the numerical simulation (COMSOL multiphysics). The theoretical equation and numerical simulation give a good correlation and it can be utilized for the inspection of the buffer block before installation. In further study, relationship between electrical resistivity and material properties (porosity and water contents) will be investigated and laboratory scale test will be conducted.

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Keywords: High-Level radioactive waste, Disposal repository, Electrical resistivity, Porosity, Degree of saturation, Water content

A Numerical Study Using OGSFLAC for the Simulation of FE Experiment in the DECOVALEX-2023

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Thermo-hydro-mechanical-chemical (THMC) coupled behavior is the key issue for securing the safety of the high-level radioactive waste disposal system. The complex interaction among the high temperature from the canister, groundwater inflow, and overburden stress makes it difficult to predict the long-term behavior in the buffer material and surrounding rocks of the disposal system. Numerous studies have been performed to investigate the interaction in various ways, including experiments numerical simulation. Laboratory and in-situ experiments are the direct methods to measure the coupled behavior efficiently, but they cost a lot for performing the tests and have difficulty in repeatability. In contrast, numerical simulation has advantages from an economic point of view and capable repeatability under various conditions and influencing factors.

Therefore, various numerical research has been performed, and DECOVALEX is one of the representative international cooperative projects for enhancing the understanding of the complex THMC coupled behavior based on the numerical simulation. The participating groups share valuable experimental data and also validate and improve the numerical methods by comparing the numerical simulation results against the test results. DECOVALEX-2023 is the current phase consisting of 7 tasks, and Task C aims to model the THM coupled behavior in the disposal system based on the Full-scale Emplacement (FE) experiment at the Mont-Terri underground rock laboratory.

KAERI is one of the participating groups in Task C and performs the numerical simulation based on the developed numerical simulator, OGSFLAC. The simulator connects the OpenGeoSys, an open-source code for the TH simulation based on the finite element method, and FLAC3D, a finite-difference method-based code for mechanical simulation, with a sequential coupling method. We adopted the Richards flow model to simulate two-phase flow systems in the buffer material and generated a simple numerical model with a domain size of 50 by 50 meters having a unit thickness.

In the numerical model, we emplaced the heater with constant power horizontally based on the FE experiment and monitored the pressure development, temperature increase, and mechanical deformation at the specific monitoring points. We monitored the capillary pressure as the primary effect inducing the flow in the buffer system, and thermal stress and pressurization were dominant in the surrounding rocks' far-field area. The results were also compared and validated with the other participating groups, and a full-scale three-dimensional numerical simulation will be continued as the following step using the developed numerical model.

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Keywords: DECOVALEX-2023, FE experiment, OGSFLAC, OpenGeoSys, FLAC3D

Integrated Operation and Management System for Geographic Information of Natural Barrier

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It is expected that geological field data and computational modeling data will be collected and produced extensively while conducting development of an assessment technology for the performance of long-term evolution of the natural barrier research. It is also clear that the amount and size of data files will increase over time due to the development of computer technology as well as the research period. Therefore, installing integrated operation and management system for vastly accumulated data is essential. In this study, the functions that can be used and to be added in the future are presented and the way to utilize are introduced. The integrated operation and management system called GDS (Geo-Data System) has three main purposes: data storage and management, sharing data between users, and managing data version. Among them, as data accumulate rapidly and sharing between users become complex, managing the final version of data is particularly important. And updating additional functions are planned to utilize GDS more systematically and conveniently. First, provision of review and feedback system on the geological field data and computational modeling data. Second, GIS-based data user interface including relative location information of each data. Third, linked viewing between related data (report, journal paper, field data, modeling data, etc.). Fourth, instant file viewing on GDS without file downloading. In addition to the above-mentioned functions, additional required functions will be added to GDS continuously in the future. After all additional functions have been applied, GDS is expected to provide advantages as integrated operation and management system to conduct development research, and to collaborating with vast amounts of data between users.

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Keywords: Natural barrier, Integrated operation, Management system, Storage server

Site Distribution of Stress Field in Deep Geological Rock With Fault Zone

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A rock mechanical site descriptive model of KURT (KAERI underground research tunnel) already had been proposed for an investigation of natural barrier behavior in a deep geological environment. The extended model with a 1 km³ site scale was constructed using integrated information of major fault and rock mass with representative properties obtaining from field data and drill core investigation.

The key approach in this study is a verification of the predictive stress regime of the model through comparison with in-situ stress from the deep depth borehole test in KURT. The aspect of changes in the stress field could be simulated with depth-dependent vertical and horizontal stress. The principle stress presented that it may vary under the influence of various geological characteristics. The significant factor includes the topographical shape of the top surface giving rise to significant variation in stress due to overburden. And also it is important to have well-defined features of rock fault and predominant rock mass properties. The stress magnitudes at monitoring locations in the model yield reasonable results for an interpretation of changeable principal stress and comparison with field test results estimated from wellbore wall failure phenomena. The ratio of horizontal to vertical stress was estimated along with the geological depth and it should be possible to assume an expected elevation of transition stress regime from reverse fault to strike-slip. The verified model for geotechnical investigation could be adopted to understand the dominant mechanisms and factors affecting the behavior of the system. Analysis for geological structures in or on-site and block scale model must be achieved with relatively a few field data. Deformability and strength properties of rock mass and fault may vary considerably so it is impossible to obtain complete field data. Therefore, for idealizing the geological model for numerical analysis, a reasonable range of determining rock mass properties was investigated. The influence of characteristics of rock fault depends on the constitutive model was also estimated.

Through this study, a strategy for a continuous spectrum between field test and modeling has been presented. It is also possible to verify the established rock mechanical model and to identify significant input factors for reliable stress field estimation. This research will extend to develop a hydraulic-mechanical coupling model and analyze the long-term evolution of natural barriers.

Keywords: KURT, Fault and rock mass properties, Model verification, Predictive stress regime

A Case Study of the Cause and Characteristics of Sulfidic Material and Gas Generation in the Underground Repository System

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Investigations were carried out on sulfate generation, characterization, and related reactions in the components of the underground repository system. Sulfate generation, characterization and related reactions in the canister are primarily related to precipitation and secondly related to corrosion of copper overpack. Dissolution and precipitation of sulfate occurs in the buffer that is associated with oxidation-reduction reactions in bentonite to form sulfate and reduce to sulfide. In the backfill, sulfate can be generated during the alteration of accessory minerals and the content of sulfate is different according to the basic characteristics of groundwater from the geosphere. In the buffer, backfills, and geosphere, the formation and reduction of sulfate can occur by microbial activity in common. Silver, the main component used in the protective film of spent nuclear fuel in the canister, can be corroded by sulfide and the sulfide gas and sulfidic material and gas can protect the oxidative dissolution of UO_2 under certain conditions. In addition, sulfide can cause corrosion of the copper overpack and can be generated and decomposed by microbial activity in the buffer, backfill, and geosphere, like sulfate.

In major overseas collaborative research cases, the projects were divided into three major themes. As the first topic, experiments were performed to measure the sulfide concentration in the geosphere and to understand the sulfidation process. As a second topic, experiments were conducted to understand the sulfide process in the buffer and backfill. Finally, integrated modeling of the safety case was performed.

Sulfide is a major material that can cause corrosion of canisters in a repository, and the mechanisms related to the formation and decomposition of sulfide are important factors in evaluating the long-term safety of underground repository system. Through this study, we can understand the interaction of microbial, chemical and hydraulic processes of sulfate and sulfide. In addition, by arranging knowledge related to the behavior of sulfidic material and gas occurring in the near-field area and geosphere, uncertainty that may be caused by sulfide in the construction of the repository can be resolved.

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Keywords: Sulfide, Sulfate, Underground repository system, Canister corrosion

Preliminary Feasibility Investigation on Long-term Field Experiments for Nuclide Migration System in KURT

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In this study the preliminary feasibility studies were conducted by conducting manufacture and installation of an in-situ nuclide migration system in KAERI Underground Research Tunnel (KURT). Radionuclides 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 radionuclides in a fractured rock is very important in the safety assessment for the radioactive waste disposal. In this study, we will describe about the manufacture and installation of an in-situ nuclide migration system for solute migration experiments in the KURT located within the research area of KAERI. The in-situ nuclide migration system was designed and installed to investigate the migration and retardation processes of solutes transported along groundwater flow through a fractured rock. The experimental system is mainly composed of two parts; 1) open system and 2) close system part. In addition, monitoring system was designed to control the pressure of a packer system. The results of the in-situ nuclide migration experiments will be used in the demonstration of radionuclide migration processes through a fractured rock and validation of a migration model for the safety assessment and its reliability improvement of the high-level radioactive waste disposal.

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Keywords: Nuclide, Migration, Long-term field experiments, KURT, Fractured rock

Wet Purification for the Selective of Montmorillonite From Bentonite

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Wet purification process for the selective separation of montmorillonite from KJ-II bentonite was studied using physical methods such as sonication, decantation and centrifugation. Bentonite clays consisting of mostly montmorillonite as a nanostructure group of smectite have been used as industrial raw material for many applications. The presence of the other impurities in bentonite clays reduces the value of bentonite. Therefore, the bentonite purification is necessary. For this purpose, a set of purification methods such as dispersion, sonication and centrifugation were used. In contrast to the other methods, there were often complex, expensive and non-effective; this purification method was found to be effective for removing almost all quartz, carbonates and clinoptilolite as the major impurities. In this purification method, the montmorillonite content was increased from 60 to 88% with the separation yield of 70%. From the results, centrifugal force and power of sonication were the important parameter in purification yield, the physical properties and the crystalline structure of the final products. The results of laser particle size distribution analysis, X-ray diffraction and Fourier-transform infrared spectroscopy results showed the decrease of impure phase without any structural changes or damages in clay framework.

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Keywords: Montmorillonite, Wet purification, Bentonite, Sonication, Centrifugation

Current Research Status on Colloid Formation and Migration (CFM) International Joint Research in Grimsel Test Site

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Underground research laboratory (URL) plays an important role to develop deep geological repository systems for the disposal of high-level radioactive waste (HLW). In South Korea, Korea Atomic Energy Research Institute (KAERI) developed a small-scale generic underground research tunnel (KURT), and finally constructed in 2015. Although the KURT is well designed to conduct in-situ tests, the use of radionuclides is strictly restricted. To redeem the situation, KAERI has joined an international joint research programme “Colloid Formation and Migration (CFM)” in Grimsel Test Site (GTS) to share the research experiment with other countries since 2008. GTS is a generic underground research laboratory in Switzerland, CFM programme aimed to improve the understanding of the processes related to the generation of colloids from the bentonite buffer of engineered barrier system and the influence of radionuclide migration by colloid formation in a fracture host rock. At present, several institutes including Nagra, DOE, POSIVA, SKB, JAEA, CRIPI, NUMO, NDA, KIT, and KAERI are joining the CFM.

In the CFM programme, various migration experiments using bentonite colloids, conservative tracers and radionuclides had been conducted as of 2013 in order to optimize the design of long-term in-situ experiment. Afterwards, a long-term in-situ migration test (LIT) had been carried out in the “tunnel packer” which is located in the middle of shear zone of GTS for 4.5 years. In this experiment, behavior of colloid and radionuclides was investigated accompanied erosion test of the compacted bentonite in a mechanical packer system. Currently, the LIT sample was finally overcored, and conducted physico-chemical characterization. This review aims to provide the information including research design, experimental optimization, data, trial and error, and troubleshooting during long-term in-situ test using colloids and radionuclides. This work will suggest the new insights and approaches in-situ colloid migration test at URL in Korea.

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Keywords: Colloid, Migration, Radionuclides, Bentonite, Long-term in-situ test

Geochemical Analysis for Classification Granitoid as Basic Geological Data From Deep Borehole

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A radioactive waste repository must be managed safely as long as it is necessary to isolate it for a geological time. Methodologies are being developed to restore and predict the evolutionary process during the geological time of the natural barriers that make up the high level radioactive waste repository. There are several proposed methods, but deep underground disposal is recognized as the most reasonable method. The deep borehole for understanding the deep underground provides an opportunity to understand the geological environment of the underground. And its detailed petrological classification therefore can be provide as a basis for future studies. Based on this, we intend to make up for the classification of the granitoid, which is the bedrock of the study area, and finally acquire the basic data of the site evolution model.

In this study, basic geochemical classification of rocks was performed using a drill core around KURT (KAERI Underground Research Tunnel). The sampling in DB-2 (~1 km) was carried out 50 m interval according to the pre-planned sampling criteria. If change or altered parts appear, additional sampling was performed. A total of 52 samples were collected and geochemical analysis (XRD, XRF) was performed. Most of samples were located in the range of granite-granite diorite. In the future, additional chemical analyzes (trace elements, isotopes and dating analysis) will be used to study the evolution of granitoid bodies around the KURT.

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Keywords: Deep Borehole, Natural barrier, Geochemical analysis, Rock classification

Unplanned Release of Rad-materials From Nuclear Facility and Groundwater Monitoring Strategy

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At both operation and decommissioning of nuclear facilities, a leakage or unplanned release of radioactive materials from the facilities should be identified to prevent the groundwater contamination because contaminated groundwater might affect the ecosystem. So, periodical monitoring of radioactivity in groundwater during the lifecycle of nuclear facilities is required. To support the design of the automated on-line groundwater radioactivity monitoring system, we suggest necessity to define the requirements of the monitoring system around nuclear facilities for early detection and warning of groundwater radioactive contamination during/after operation and decommission. Unfortunately, there is currently no independent standard that lays out the characteristics of such an on-line integrated surveillance system. On-line integrated surveillance system developed in this study for the monitoring of groundwater radioactivity includes variable functions such as sampling, estimation, analysis, radiological identification, long-term monitoring, data storage, transmission, user control, etc., and has suggested to the International Electrotechnical Commission (IEC) as a new standard. This standard might be applied to environmental surveillance for detection of leakage or unplanned release of radioactive materials from nuclear facilities to groundwater as part of the environment, and monitoring wells, which are necessary to monitor radioactive materials that could exceed predefined limits (background levels) in groundwater by leakage and unplanned release.

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Keywords: Nuclear facility, Groundwater, Radioactivity, On-line monitoring

A Review of Measurement and Estimation Methods for Potential and Actual Evapotranspiration in South Korea

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In order to better predict the dynamics of water flow and solute transport and thus ensure the long-term safety and stability of a high-level radioactive waste repository, it is important to understand hydrologic feedback processes within a water cycle such as precipitation, interception, runoff, infiltration, evapotranspiration (ET), and recharge. For instance, when precipitation occurs, some is blocked by vegetation, and the other falls to the ground and causes direct runoff from the surface and penetration into the underground. The water that penetrated into the underground increases soil moisture contents and some recharges groundwater. Of six hydrologic processes, ET represents a water transport process from soil and vegetation to atmosphere, and accounts for 60% of the total precipitation that reaches the land surface. Since many climatic and biological factors affect ET, the measurement and estimation of ET are challenging, especially at large spatial scales. Therefore, various methods have been developed for the evaluation of ET, including Bowen Ratio, Eddy Covariance, Optical Scintillation, and weighing lysimeter methods. In this study, a comprehensive review of ET assessments in South Korea is presented. By comparing the advantages and limitations of ET evaluation methods, several methods for ET assessments around KAERI underground research tunnel (KURT) are identified and discussed.

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Keywords: Safety, Water cycle, Hydrologic process, Evapotranspiration, KURT

Review of Alternative Disposal Concepts for Spent Nuclear Fuels

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Most countries, which utilize the nuclear energy, confront with the problem of how to manage spent nuclear fuels (SNF). Some prefer to direct disposal of them, and others prefer to recycling of the fissile materials within them. In 2016, our government announced the National Plan (NP 2016) regarding the final management of high-level waste (HLW) in ROK, but there have been still many oppositions to the government decision from NGOs. In 2019, new committee were organized to review the NP 2016, and they submitted the final recommendations to the government in 2021. So far, KAERI has developed the technologies related to the final management of SNF in two directions, that is, direct disposal based on the 'KBS-3 concept' and recycling based on 'Pyroprocessing-and-SFR'.

Authors proposed 'a strategic approach' for the future National Plan of the management of SNF. The main idea for this approach is that the government should choose the best option available at the moment when the decision is made and that this generation has no right to determine the permanent method for the management of spent nuclear fuel. Therefore, our role is to develop many options for the management of SNF for that decision. Within the ROK, currently, the most probable disposal option is KBS-3 type geological disposal, but no one knows what the best option will be in 20 or 30 years.

In this work, we review all the alternative disposal concepts focusing on the direct disposal of SNF and compare the pros and cons of them. Total nine different concepts including the reference KBS-3 concept are compared, and their technology readiness levels (TRL) are evaluated according to US DOE/EM classification. Among them, two concepts, sea dumping and deposit under ice-sheet, are prohibited currently because of the international convention and treaty. Also, it is believed that it requires more scientific or engineering development to make 'launch into space' a reality.

Finally, five disposal concepts-① disposal in rock under sea ② indefinite storage and disposal (CARE repository) ③ very Deep Borehole Disposal ④ Mined deep Borehole Matrix ⑤ Deep Horizontal Borehole- are selected as candidate concepts. Among the five alternatives, we will select the alternative concept for CANDU and PWR spent nuclear fuel, respectively, through a comparison study with the reference concept, KBS-3, next year.

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Keywords: Spent nuclear fuel, Alternative disposal concepts, Long-term storage, TRL

A Preliminary Investigation on Geochemical and Mineralogical Characteristics of Ogcheon Uranium Zone for Natural Analogue Study

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A preliminary investigation on geochemical and mineralogical characteristics of the Ogcheon uranium zone is carried out for a natural analogue study using uranium depositions. The Ogcheon uranium zone is a representative sequence of areas including uranium deposits of relatively high quality in the middle of Korean peninsula. Major geological formations containing uranium in the Ogcheon uranium zone is found to be carbonaceous black slates of the middle to lower Guryongsan formation and its equivalents. Uranium rich zones are found in Chungju, Deogpyong-Yongyuri, Miwon-Boun, Daejeon-Geumsan areas with a range of 0.017–0.087% uranium content. Major minerals of uranium-bearing black slates are identified as quartz, sericite, biotite and chlorite. According to the chemical composition of the black slates, the content of Al_2O_3 was much lower than average values suggesting Changri basin has rather proximal to its source area.

Trace elements closely correlated with uranium content is estimated as V, Mo, Ba, Ni, Zr, Pb, Cr, and fixed carbon by chemical analyses for black slate and lime stone of the Ogcheon uranium zone. Estimated pH (7.0) and Eh (−0.3 V) values for the black slates in the middle of the Ogcheon uranium zone suggest that uranium was deposited under an organic-rich anaerobic marine environment. It is also expected that decomposition of organic matters to H_2S under reducing conditions can reduce the uranyl ($\text{U}^{\text{VI}}\text{O}_2^{2+}$) into insoluble uranium dioxide ($\text{U}^{\text{IV}}\text{O}_2$), which will resultingly contribute to the mineralization of uranium. The geochemical and mineralogical data for the Ogcheon uranium zone will be used in both siting a natural analogue study site and carrying out researches for long-term behaviors of uranium in subsurface environments analogous to high-level radioactive disposal environments.

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Keywords: Natural analogue study, Ogcheon uranium zone, Uranium, Geochemistry, Mineralogy

Corrosion Behavior of 5-year-old Copper Specimens in a KURT Aerobic Environment

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A copper corrosion module that simulates engineering barrier system have been stored in an aerobic KURT environment for 5 years, and was recovered and dismantled recently. The tested copper specimens were fabricated by a cold sprayed coating technique. The specimens were buffered by compressed Gyeongju bentonite (KJ-I) and exposed to KURT groundwater at 70°C for a long time. Recovered copper samples showed that there were white mineral deposits on the surface. The average corrosion thickness was measured by the weight loss method and was 2.3 μm higher than expected. The main corrosion product identified through XRD was Cuprite. The surface roughness of the sample was measured by Alpha Step, and the thicknesses of the mineral precipitation layer and the oxidation layer of the sample were confirmed by focused ion beam (FIB). Meanwhile, the diffusion depth of copper ions in the buffer material was analyzed by ICP-OES, and it was confirmed that the penetration depth of copper ions was about 2 mm.

In conclusion, it was estimated that the reason why the corrosion thickness was higher than expected was because the copper ions in the pore water reacted with the dissolved minerals and precipitated, which results in lowering the copper ion concentration in the void water.

Keywords: Copper, Corrosion, KURT, Aerobic, Underground, Bentonite

Review of the Foreign Status of Spent Nuclear Fuel Repository

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For safe management of spent nuclear fuel, spent nuclear fuel should be delivered to repository and disposed. Currently, there is no spent nuclear fuel repository in Korea. In order to secure a spent nuclear fuel repository, it is necessary to investigate the status of spent nuclear fuel repository in foreign major leading countries. In this study, we investigate the status of spent nuclear fuel repository in foreign major leading countries.

In the U.S, the Nuclear Regulatory Commission (NRC) is in charge of disposal regulations for spent nuclear fuel, and the Department of Energy (DOE) is the direct disposition entity. In 2002, Mt. Yucca was selected as the site for a spent nuclear fuel repository. However, in 2009, the U.S stopped the plan and formed the Blue Ribbon Committee to establish an alternative to spent nuclear fuel management. In 2017, DOE promoted the Mt. Yucca repository licensing project again. In Sweden, the Swedish Radiation Safety Authority (SSM) is in charge of disposal regulations for spent nuclear fuel, and the SKB is the direct disposition entity. The SKB selected site in four stages: 1) comprehensive location survey, 2) feasibility survey, 3) site survey, and 4) detailed characteristic survey. Through this process, in 2009, Forsmark was selected as the site for the spent nuclear fuel repository. In Finland, STUK is in charge of safety regulations for radiation and nuclear power, and the POSIVA is the direct disposition entity. The site survey was conducted through three types of surveys: 1) site confirmation survey, 2) rough site characteristics study, and 3) detailed site characteristics study. Through this process, Olkiluoto was selected as the site for spent nuclear fuel repository. The construction of the repository started in 2016, and it is planned to operate from 2023.

In this study, we investigated the status of spent nuclear fuel repositories in foreign major leading countries. The U.S, Sweden, and Finland have secured spent nuclear fuel repository sites. In addition, Finland has obtained license to construct spent nuclear fuel repository. The results of this study can be used as basic data for the safety management of spent nuclear fuel.

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Keywords: Spent nuclear fuel, Disposal, Spent nuclear fuel repository, DOE, SKB, STUK

Research Plan to Secure the Input Data for the Safety Assessment of Deep Geological Disposal Based on the KURT

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Safe and permanent disposal of radioactive waste including spent nuclear fuel is a very important and sensitive issue for the countries with nuclear power plant. Safety assessment should be preceded to ensure long-term safety for deep geological disposal. The 'General Standards for Deep Geological Disposal Facility for High-Level Radioactive Waste' announced by the Nuclear Safety and Security Commission specified that the safety assessment should be conducted for 10,000 years after disposal. For the safety assessment of the disposal system, it is necessary to secure the input parameters for the migration and retardation of radionuclides considering the evolution of the disposal system.

Evolution-related factors affecting the geochemical behaviors of radionuclides include changes in redox potential, incorporation of sea water, temperature changes, and formation of hyper-alkaline conditions due to the dissolution of auxiliary components, which may occur in the disposal system after the closure of the repository. The sorption and diffusion are the main mechanisms that can retard the migration of radionuclides from a deep geological repository to the ecosystem. Therefore, we have established a research plan for the sorption and diffusion of radionuclide based on the KURT (KAERI Underground Research Tunnel), which is a virtual disposal site.

Keywords: Diffusion, Disposal, Radionuclide, Safety assessment, Sorption

Analyses on the Decay Heat Management in a Disposal Canister of Deep Geological Disposal System for Spent Nuclear Fuels

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High-level radioactive wastes including spent nuclear fuels produced by the nuclear power industries must be disposed of in a way which assures that radionuclides will be safely and permanently isolated from the human environment. Currently, the most widely agreed disposal method is to dispose of spent nuclear fuels in a carefully designed underground repository constructed at the depth of around 500 m in a selected geological formation. The safe disposal of the radioactive waste depends on the performance of the overall waste disposal system composed of engineered barrier and natural barrier.

A main consideration in both barriers is the heat generated by the waste. The heat will eventually flow from the waste form, through disposal container and buffer to the surrounding rock. A systematic assessment of the thermal loading of the repository is required to assure that the heat from the wastes will have a controllable effect. This includes studies of the effects of heat on the bentonite buffer, disposal containers, structural stability of the host rock and radionuclide release and migration from the conditioned waste forms to the environment. That should lead to establishing maximum acceptable thermal loads and maximum temperature for underground repository components that will not violate safety, engineering, and operational requirements.

Generally, for the engineered barrier, the thermal constraint has been specified to keep the integrity of bentonite buffer and to avoid the boiling of water near the disposal container. This is specifically to avoid excessive drying of the bentonite and to limit the corrosion of the copper canister because of the enrichment of potentially corrosion-accelerating salts at the canister surface. There are several options to manage the thermal load in the disposal canister. The options can be the control of waste container size, tunnel and/or waste container spacing, cooling time of the waste and the thermal conductivity of buffer and backfill, etc.

In this paper, the thermal constraints for various deep geological repository concepts and the thermal load management for deep geological disposal in domestic and abroad were reviewed. And, to consider as an input parameter for the effective disposal system, among the options for the thermal load management, the controls of decay heat capacity of a disposal canister for spent nuclear fuels in the KBS-3 type repository were analyzed.

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Keywords: Spent nuclear fuel, Geological disposal, Thermal constraint, Thermal load management

Gamma Radiation-induced Corrosion of Pre-oxidized Cu in De-aerated Pure Water

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In a deep geological repository, Cu canister is the primary engineered barrier for confining radionuclides in used nuclear fuel. Predicting long-term integrity of Cu canister requires a deep understanding of complex multi-physics phenomena without undisclosed physical changes and chemical reactions. Despite decades of the previous studies, radiation-induced phenomena are fully understood. This study aims at analyzing the effect of varying Cu oxide layers on corrosion in de-aerated ultrapure water under varying gamma-rays dose. Pre-oxidized Cu oxide layers mimics initial oxide layers of Cu canisters formed during fabrication, storage, and transportation. Also, there are oxygen gases and ions in engineered barrier systems at the early phases of geological repository. Polished Cu metals are pre-oxidized at 90 and 140°C. The two different sets of samples were irradiated in de-aerated ultrapure water by gamma cell (^{137}Cs , $\sim 0.2 \text{ Gy}\cdot\text{s}^{-1}$). Total radiation dose varies at 40–200 kGy, roughly equivalent to accumulated dose for 100 years of gamma-rays irradiation from used nuclear fuel. After irradiation, the amounts of Cu oxides on Cu samples and in pure water were respectively measured by the cathodic reduction method and ICP-OES (Inductively Coupled Plasma Optical Emission Spectroscopy). The thicker Cu specimens as well as higher radiation exposure doses tend to produce more accelerated corrosion rates, however the amounts of corrosion products are overlapped within standard deviation. To understand such results analytically, a numerical model was constructed in the COMSOL Multiphysics to simulate the most corroded Cu sample case in the experiment. The model showed that O_2 was a major oxidant species under gamma-rays irradiation, which agrees to the surface analysis of irradiated Cu samples.

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Keywords: Gamma rays, Geological repository, Copper canister, Irradiation

A Simulation Study to Evaluate the Thermal Expansion Behavior of Dilatometer Test

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A geological repository system consists of a disposal canister with packed spent fuel, buffer material, backfill material, and intact rock. Among these, the buffer is one of the most important components to assure the safe disposal of high-level radioactive waste (HLW). Since the thermal expansion phenomenon is subject to cause thermal stress to canisters and intact rock mass, it is required to evaluate and examine the thermal expansion behavior of the buffer in advance from the design stage of the system. In this study, a numerical simulation model was proposed to evaluate the thermal expansion behavior of KJ bentonite buffer for the dilatometer test. Thus, the linear thermal expansion coefficient of the dried and saturated bentonite was calculated with varying the initial porosity and temperature and specific gravity. According to the simulation results, the linear thermal expansion coefficients were in the range of $3.5\text{--}7.0 \times 10^{-6}/\text{K}$ and it has positive correlation with initial porosity and specific gravity. Furthermore, based on the database from the simulation results, the empirical formula could be made for estimating the linear thermal expansion coefficient of compacted bentonite by quantitatively examining this correlation.

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Keywords: Numerical simulation, Bentonite buffer, Dilatometer test, Thermal expansion coefficient

Preliminary Study of Classification of Geological Risk Factors for Deep Underground Space in Korea

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According to examples of rock mass collapse that occurs when excavating deep undergrounds around the world, (1) geological structures such as faults, fault clays, crushed zones, folds, dykes, etc. as the cause of tunnel collapse and deformation; (2) rock properties such as weathering degree, joint spacing and roughness, type of rock, uniaxial compressive strength, strain coefficient, RQD, Q value, alternation of limestone, anisotropy, etc.; (3) high stresses such as differential and principal stress states, rock burst, squeezing, spalling, slabbing, etc.; (4) hydrogeology such as inflow of water and mud, groundwater level fluctuation, pore water pressure, rainfall intensity, etc.; (5) artificial structures such as tunneling method, the cross-sectional shape of the tunnel, characteristics of the support material, the cross-sectional area of the artificial cavity, the degree of damage to the support material and surrounding buried structures, over-excavation, etc.; (6) Overburden such as tunnel depth, overburden load, strike/dip of soil and rock boundaries, etc.; It is divided into (7) ground characteristics such as karst and topography, geothermal heat, earthquake, ground displacement due to InSAR, etc. In other words, the risk factors of rock mass excavated from 100 m or more to several kilometers deep below the ground are classified into about 30, and can be divided into 7 categories. Using these risk factors as basic data, weights for each factor for each category can be set, and further, the risk of excavated rock mass can be calculated.

Keywords: Risk factor, Geological structure, Rock properties, High stress, Hydrogeology, Underground space

Enhanced Iodine Sorption by Layered Double Hydroxides Modified Bentonite

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Radioactive iodine is one of the major concerning radionuclides due to its high volatility and radiotoxicity. In addition, iodine has high solubility, so it can dissolve easily in groundwater and transport to the environment. Therefore, a novel method should be developed to limit the iodine mobility from the perspective of radioactive waste repository safety performance and long-term risk assessment.

Bentonite, which is classified as 2:1 clay like smectite, mainly consists of montmorillonite and is used as an engineering barrier in the repository. Montmorillonite has a central octahedral alumina layer between two tetrahedral silica layers, and partial of aluminum and silicon are substituted by lower valence metals. Charge imbalance caused by isomorphous substitution is neutralized by the insertion of exchangeable cations. Water molecules easily adsorb on cation prevalence interlayer site and increase the volume of bentonite. The swelling properties of bentonite can also prevent the penetration of groundwater in the repository and minimize the release of radionuclides. Bentonite is well known to remove cationic radionuclides including cesium by ion exchange with intercalated cation, however rarely react to anionic radionuclides such as iodine and technetium due to the permanent negative surface charge.

Layered double hydroxides (LDH), known as anionic clay, has positively charged, inorganic layered compounds with exchangeable interlayer anions. LDH is similar to brucite-like layer structure and partial of divalent metal ions are substituted by trivalent cation, resulting in positively charged layers. The charge imbalance is compensated by the intercalation of exchangeable anion between the layers. The anion exchange properties of LDH are useful to eliminate anionic radionuclides such as iodine. To improve the shortcomings of bentonite, we modified the surface of bentonite using the formation of LDH (LDH-Ben) to increase the removal capacity of bentonite for iodine.

Various LDH-Ben have been developed by modifying the divalent cations and used for batch sorption tests at solid to solution ratio 0.4 g/40 mL containing $100 \text{ mg} \cdot \text{L}^{-1}$ of I^- or IO_3^- . Original bentonite did not remove iodine(both removal efficiency of I^- and IO_3^- are 0.0%) from solution, however LDH-Ben showed the increment of iodine sorption(removal efficiency of I^- ranges from 18.4 to 35.0%, and that of IO_3^- ranges from 92.1 to 98.5%). In particular, all LDH-Ben showed higher removal efficiency in IO_3^- than I^- . Further experiments are planned to elucidate the iodine removal mechanism of LDH-Ben.

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Keywords: Layered double hydroxides, Bentonite, Repository, Iodine

Analysis of Correlation Between Climate Change and Long-Term Evolution of Surface Environment Using FEPs

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Long-term time scale should be considered for evaluating deep geological disposal due to long-lasting risk of the radioactive wastes. During these long-term time scale, the surface environment could be changed. The changes of surface environment can have influence on safety assessment by changing transport path of radionuclides. It may also cause changes in the other parts of the disposal system.

For describing temporal changes of the disposal system, including the biosphere, and its components, it is required to understand features of specific components or objects, events that are likely to happen over a long time, and processes that interacts among features or change the features by the events. This is the reason that features, events, and processes (FEPs) list is necessary. When the scenario representing long-term evolution of disposal system is formulated, FEPs that are directly involved in the scenario and interaction among the FEPs are very important.

There are many factors that cause long-term evolution of the surface environment. In order to stylize long-term evolutions of the surface environment and formulate scenarios using these factors, the correlation between climate change and the resultant changes in the surface environment factors must be identified. If FEPs indicating the status and changes of the disposal system has been prepared, it is necessary to clearly suggest the procedure for representing the long-term evolution in the surface environment due to climate change by identifying the interaction or correlation between the related FEPs.

In this study, in order to present a method or procedure to explain the long-term evolution of the surface environment, including the biosphere, caused by climate change using FEPs, the FEPs representing the components of climate change and the surface environment was evaluated. And evaluations of the correlation between each FEP was attempted. For this purpose, FEPs related to climate change and surface environment components that are determined to be used to represent long-term evolution of the surface environment due to climate change were selected, and the correlation between each FEP was roughly presented.

Keywords: Long-term evolution, Surface environment, Climate change, Geological disposal, Radioactive wastes

Investigation of the Long-term Evolution of the Biosphere for the Safety Assessment of a Deep Geological Repository

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Securement of a safety is a top priority issue during the development of a deep geological repository for the disposal of spent nuclear fuels. The safety assessment of a radioactive waste repository is made for the radionuclide release scenarios and the biosphere assessment. The radionuclide release scenarios are developed by taking account of all FEPs (Features, Events, and Processes) related to the disposal system including biosphere. In general, the timescale for the post-closure safety assessment ranges from ten thousand to a million years. Therefore, the long-term environmental change is recognized as an important issue in the post-closure safety assessment. The major drivers of environmental change are the climate change and the resulting changes in hydrology and landscape. In this study, we investigated the long-term evolution of the biosphere by making FEP list of the biosphere considering the climate change and the landscape development. We classified the main categories of the FEPs into a biosphere and an external factor. The biosphere is again classified into a surface environment and a radiation exposure. And the external factor is classified into a climate change, a landscape development. We make a list of FEPs for these sub-categories of the FEP related to the biosphere. Each FEP is described using a standard format including a FEP name, a type, a class, a general description, a linked FEPs, a long-term evolution, a time frame, and references. The resulting FEP list will be used for the formulation of radionuclide release scenarios considering a long-term evolution of the disposal system.

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Keywords: Safety assessment, Long-term evolution, Biosphere, Climate change, Landscape development

Physical and Chemical Properties of Bentonite on the Interaction Between Different Corrosion Products and Buffer

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The spent fuel disposal canister is one of the components of an engineering barrier for deep disposal and is the final stage of separating spent fuel from the external environment. Corrosion reaction in the disposal canister changes their mechanism and corrosion products as the disposal environment evolves, and the resulting corrosion products may remain in the form of sediments at the interface between disposal canister and buffer or penetrate into buffers by transformation, resulting in changes in buffer performance.

The purpose of this study is to understand physical and chemical properties of buffer in contact with a copper canister being developed as a high-level radioactive waste disposal canister. In this study, it was intended to conduct a reaction experiment and evaluation of Ca-bentonite (Gyeongju bentonite, KJ-II) and Na-bentonite to be used as a buffer material in contact with rolled plate, cold spray coating copper (CSC), SUS 304, cast iron and Ti products. We investigated density, swelling index, element concentration and chemical composition of the bentonite using mineral oil, KS K 0764, ICP and XRF.

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Keywords: Buffer, Bentonite, Corrosion, High-level radioactive waste, Physical and chemical properties

An Example of Application of Natural Analogues to Geologic Repository Programs, Focusing on the Studies Have Been Used by the Yucca Mountain Site

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For the Yucca Mountain Site characterization project, natural analogues have been applied in quantitative or qualitative ways depending on the purpose to which they were applied and on the specific analogue. They provide descriptive information about the occurrence of various processes or qualitative information of those processes. In Yucca Mountain Site characterization project, natural analogues allow the testing of individual processes over geologic time and space scales, assessing the relative importance of various processes coupling. Especially, for some processes (e.g., those that are thermally coupled), natural analogues may be the only means of providing the required understanding of long-term and large-scale behavior needed to provide scientific confidence in process models for input to total system performance assessment (TSPA) according to their reports.

Natural analogues have to be considered in terms of repository design variables rather than attempting to match a specific design because of the flexible design approach of Yucca Mountain Site. The important variables are temperature of drift walls, presence or absence of a boiling front, dimensions of drifts, absence of backfill, and ventilation. Because of the flexible design approach of Yucca Mountain Site. These variables affect drift stability, waste package and drip shield corrosion, processes operative within the engineered barrier system, and thermally coupled processes affecting the host rock at the drift and mountain scales. In this manner, natural analogues are used as a means of model validation or confidence-building.

In this study, the application of natural analogues to the Yucca Mountain Site Project was to get reviewed to understand the role of natural analogues and their different approaches, if any, to the geologic repository program compared with other countries.

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Keywords: Natural analogue, Yucca mountain site, Geologic repository program, Repository design

Natural Analogue Study on Disposal of Radioactive Waste and Study on Concrete Deterioration

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Natural analogue studies are underway to predict the possibility of dissolution of radioactive wastes and flow of groundwater in the process of disposing them. They are defined as a study of the natural environment which has some similarities to a disposal site, such as matters and the surrounding environment, to protect the wastes placed there against external deterioration. However, it is difficult to precisely carry out a natural analogue study due to various deterioration phenomena, such as settlement, earthquake, or movement of groundwater.

Natural analogue studies on the performance of the major materials used to protect radioactive wastes from external deterioration are being conducted. One of the major materials is concrete, which plays an extremely crucial role in the disposal of such wastes. The studies on concrete may show large gaps in its durability due to various deterioration factors, such as interactions with groundwater and other substances. Some of the key factors are deterioration triggered by sulfate attack and a repeated cycle of its freezing and thawing.

Concrete deterioration induced by sulfate is commonly manifested in an extensive scale when a concrete structure makes contact with soil or water, aggravating its performance. In this case, the penetrating sulfate generates ettringite, which has a larger volume than the material, leading to abrasion of the concrete and occurrence of cracks due to its expansion.

Concrete deterioration caused by freezing and thawing (freezing damage) refers to a repeated process of freezing and thawing due to the penetration of groundwater, which causes cracks, flaking and damage on the material.

This preliminary study analyzed the necessity of natural analogue studies and the factors contributing to concrete deterioration. Those studies are essential in protecting radioactive waste at a disposal facility from external deterioration, and it has been analyzed that concrete is a major material to shield the wastes against the phenomenon. The contributing factors include sulfate attack and alternative repetition of freezing and thawing. However, there are not sufficient natural analogue studies on concrete deterioration, requiring further research on the subject to keep radioactive wastes from external deterioration.

To further analyze the characteristics of concrete deterioration, a relevant experiment is expected to be carried out where a cement mold is cast and sulfate solution is used to penetrate into the container after freshwater that reflects the groundwater conditions.

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Keywords: Concrete, Deterioration, Natural analogue study, Sulfate attack, Freezing-thawing

Theoretical Study on the Reaction of Xenon and Fluorine From DFT Calculation

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The behavior of radioactive gases like radon (Rn-222) in the underground environment is very important for the assessment of radiological risks to the public. As deep disposal of spent nuclear fuel, in case of damage to container and loss of the surrounding natural barrier, radium-226 (the parent nuclide of Rn-222) is generated as a decay product after 4,000 years. Therefore, it is necessary to study the behavior of Rn-222 and its solid phase.

In this paper, we investigated to find the optimal DFT (Density Functional Theory) calculation conditions for xenon (Xe) and fluorine present in the solid phase. The optimal conditions obtained from this study are applicable to the relationship between radon and fluorine, because Rn and Xe are the same periodic elements.

Periodic calculation was performed with the Vienna Ab initio Simulation Package code (VASP 4.5). In the VASP program, the Kohn-Sham equations for exchange-correlation energy are solved with various methods such as LDA (Local Density Approximation) and GGA (generalized gradient approximation). To optimize the geometry structure of xenon fluorides (XeF_2 , XeF_4 and XeF_6), the bond length and bond angle were compared using GGA and LDA. The calculation results of XeF_2 and XeF_4 showed that both LDA and GGA provided bond lengths with less than 5% error from the experimental values and the bond angles were precisely consistent with 180° and 90° , respectively. The XeF_6 has been controversial with regards to the stereoactivity of its valence electron lone pair. Our calculations confirmed that XeF_6 has an O_h symmetry. Experimental evidence has shown that XeF_6 is closer to C_{3v} , not O_h symmetry. Based on the Xe-F calculation results, same phenomenon is expected to be for the case of radon hexafluoride (RnF_6) as well. Additional studies are needed in the structural optimization of XeF_6 , which can be applied for Rn behavior in the long-term repository.

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Keywords: DFT (Density Functional Theory), VASP, Xenon, Fluorine, Radon

Evaluation of Backfill Compressibility in the Disposal Repository for Performance Requirement

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The backfill is key part of the Korean high-level waste disposal repository along with the disposal canister and buffer and plays an important role in preventing leakage and movement of high-level waste. SKB and Posiva have established mechanical, hydraulic, and geochemical performance requirements to contribute to protecting the container and buffer. Hydraulically, to prevent the groundwater flow, hydraulic conductivity should be less than $10 \text{ m}\cdot\text{s}^{-1}$, and for mechanical performance, a swelling pressure of at least 0.2 MPa is required to restrain upward movement of the buffer during saturation and swelling. In addition, backfill requires the compressibility to maintain the saturated buffer density at top of the canister over $1,950 \text{ kg}\cdot\text{m}^{-3}$ with sufficient margin. Performance requirements for backfill can be achieved by feasibility evaluation after defining concept, preliminary experiment, and in-depth test that will be the basis for selection of candidate materials and backfilling methods. Advanced countries for disposal research have had various experiences and achievements in research on characteristics of backfill candidate materials. However, in Korea, research on the function and performance of backfill is lacking and has been started recently. In this study, an analysis of the mechanical performance criteria requirements for establishing backfill was performed. A theoretical background of backfill that can calculate the saturated buffer density during swelling was established. Based on the theoretical background, the compressibility characteristics of the backfill to meet the performance requirements were estimated. This study can be used to evaluate the performance of various backfill material candidates.

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Keywords: Disposal repository, Backfill, Compressibility, Performance requirement

Analysis of Design Requirements for the Geological Repository for Spent Fuel

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A geological repository for spent fuel would be constructed in the hard rock at a depth of several hundred meters below the ground surface. After the emplacement of the canister containing spent fuels into the deposition hole, the buffer, backfill, plug and closure are installed and the geological repository is closed. Every country promoting the construction of a geological repository for spent fuel has its unique disposal site conditions, and there are also differences of the characteristics of spent fuel to be disposed of, the national policy, and the regulation system for the geological disposal among the countries. Therefore, the design requirements adopted for the geological repository are different from country to country. This study is intended to analyze the repository design requirements that have been developed in the country and abroad, and to suggest the design requirements suitable for Korean situation. The state of the art and the related research results on the barrier function, the performance requirements, and the design requirements of the barriers for the geological repository are analyzed. For each barrier in the geological repository, the barrier function is defined and the qualitative performance requirements to meet the barrier functions are established. Based on the performance requirements established, the various research and measurement results are analyzed and assessed. The quantitative design requirements for each barrier of the geological repository are suggested. However, the several design requirements suggested are described qualitatively without the quantitative values because of their nature. The function, the performance requirements, and the design requirements for the barriers in the geological repository established through this study could be useful to develop the design technology and the related technologies for Korean geological repository. However, these performance requirements and design requirements should be modified and improved in the future considering the characteristics of the disposal site, the repository concept, and the regulatory requirements. Also, a great variety of the research, development, and the demonstration of the related technologies are required for the development of more accurate design requirements suitable for the Korean situation.

Keywords: Geological repository, Design requirements, Performance requirements, Barrier function

A Review on Effects of Hyper-alkaline Alteration of Buffer on Migration of Radionuclides Under Repository Conditions

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The goal of the deep disposal of high-level radioactive waste (HLW) is to permanently isolate the HLW from human life and the ecosystem. The conceptual repository model for deep disposal consists of multi-barrier system to block the exposure of radionuclides to the ecosystem. Multi-barrier system consists of an engineered barrier system including canister, buffer material, and backfill material, and a natural barrier system including the bedrock near the repository and its surrounding underground environment. The buffer material in the engineered barrier protects the canister from the surrounding physical impact, minimizes the inflow of groundwater through the bedrock near the repository, and retard the release of radionuclides over a long period of time. For effective performance as buffer, the buffer should have low permeability, high adsorption capacity, self-sealing capacity, and high thermal conductivity. However, these performances of buffer can change with the evolution of the repository environment, which can seriously affect the migration of nuclides. In this study, case study on the effects of hyper-alkaline alteration of buffer on migration of radionuclides under repository conditions was introduced. The loss of performance of the buffer due to the hyper-alkaline alteration occurs through the geochemical reaction between infiltrated groundwater and a large amount of cement material used in the construction of the repository. After interactions between the alkaline solution and buffer, the migration of the radionuclide can be accelerated with processes such as cation exchange, dissolution-precipitation, and illitization. Also, this process can lead to decreasing retardation capacity and swelling pressure, and increasing hydraulic conductivity of the buffer. Therefore, evaluating the long-term safety of the buffer under hyper-alkaline conditions is one of the important factors in predicting long-term migration of radionuclide after repository closure.

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Keywords: Hyper-alkaline alteration, Buffer, Migration, Radionuclide, Repository

A Stepwise Safety-Case-Based Regulation System for Deep Geological Disposal of High-Level Radioactive Waste

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A regulation system has been designed based on the concept of safety case to guide the domestic development of the deep geological disposal system for high-level radioactive waste through a systematic, stepwise procedure.

The design starts from defining what is the safe disposal and how it can be realized, which should be the context of safety case, including the safety objectives and strategy for their achievement. The disposal safety could be assured with concert of three pillars of effort: safety system construction, the relevant system development, and their systematic integration with the safety case throughout the development. The safety system construction means to comprise the required safety functions in depth by isolation and containment of radioactive waste, which include environmental safety features for both the present and the future, the natural barrier's safety features (such as geological stability, hydrogeological characteristics, and geochemical properties), the safety features of engineered barriers (such as spent fuel canisters or waste containers, buffers between the canister and the disposal hole, and other structural elements), and the safety features of waste form itself. The system development is the process preparing and embodying the relevant repository, which may be well arranged in three groups: site development (i.e., basic survey, detailed investigation, characterization, and overall management), on-site development of the underground research laboratory (i.e., construction and operation), and development of the disposal facility (i.e., design, construction, operation, closure, and post-closure control). The facility design should particularly consider waste acceptance, system monitoring, waste retrievability during the facility operation, and nuclear security. Most importantly, the two pillars mentioned above, safety construction and the system development, should be integrated with the stepwise safety case through the development process: the generic phase with basic system study, the site-specific phase beginning with site survey, and the facility-specific phase with practical development of the disposal facility, which may be subdivided into design, construction, operation, closure, and post-closure phases. In particular addition, a pertinent management system, which is also a component of the safety system, should be employed as an essential element to the safety case development.

The regulation system requires each safety case be constructed in accordance with the purpose of development at the stage and the correspondingly up-to-date information on the safety construction and the system development. It also needs establishing relevant laws and ordinances to the system development in response to the safety requirements and standards. This regulation system is a basis for preparing the stepwise, specific regulations and the compliance criteria to the three pillars for the deep geological disposal system development, and it is applicable to the safety case as a basic guidance.

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Keywords: Deep geological disposal, Regulation system, Safety case, High-level radioactive waste

Korean Bentonite Buffer Characterization Under the Elevated Temperature Conditions for SNF Disposal

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In most countries, the upper limit of buffer temperature in the repository is set to below 100°C due to the possible illitization. This smectite-to-illite transformation will likely be detrimental to the swelling functions of the buffer. However, if the upper limit increases, the disposal density and cost-effectiveness for the repository will dramatically increase. Thus, understanding characteristics and providing database related for the buffer under the elevated temperature conditions is of crucial importance. Here, we review and propose a strategy to investigate Korean bentonite under the elevated temperatures, from a mineral transformation and radionuclides retardation perspective. The long-term hydrothermal reactions provide bentonite samples required. The bentonites will be studied using *in-situ* synchrotron-based XRD technique to determine smectite-to-illite transformation. The ‘high-temperature and high-pressure mineral alteration measurement system’ based on Diamond Anvil Cell (DAC) will control and provide the elevated temperature and pressure conditions simultaneously during the measurements. Kinetic modellings will predict how and when the illitization becomes detrimental to the performance and safety of the repository. The sorption reactions planned for the bentonites to evaluate effects on retardation will provide the information required to expand our knowledge for repository optimization.

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Keywords: Korean bentonite, Elevated temperature conditions, Smectite-to-illite transformation, Radwaste disposal

Inorganic Surface Modification of Bentonite to Increase Radionuclide Removal

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Bentonite is a material to be used as a buffer in the engineering barrier system of a deep geological repository due to its swelling property and high sorption ability. In particular, cation sorption ability is a factor directly related to the role of preventing or delaying the potential leakage of radionuclides from canisters. Thus, this research focused on improving the radionuclide sorption ability through surface modification (or transformation) of bentonite.

To date, there have been several studies that have improved the radionuclides sorption ability by modifying the surface of bentonite with an organic surfactant, such as hexadecyl trimethyl ammonium, polyvinyl alcohol, and sodium carboxymethyl cellulose, etc. However, considering that bentonites to be used in the deep geological repository have the limited organic content, surface modification using organic surfactants is virtually impossible to apply.

Here, we established a surface modification method without using any organic substances. Using sodium metasilicate, sodium aluminate, and sodium hydroxide, we tried to build a zeolite structure on the surface of bentonite particles. Although the diffraction pattern of montmorillonite (major mineral phase in bentonite) was decreased and shifted as original Ca-bentonite was ion-exchanged with Na-bentonite during the surface modification process, it was confirmed that spherical NaP zeolite clusters were well-formed on the surface of the bentonite.

During the surface modification, when the amount of initial bentonite used was small, NaX zeolite not NaP zeolite was also found. When the same amount of aluminosilicate gel prepared from the inorganic additives is present without bentonite as an initial substrate used, NaX zeolite can be produced only at the homogeneous nucleation site inside the gel, which provides enough gel per site for NaX zeolite formation. On the other hand, using with bentonite, zeolite can be produced not only at the homogeneous nucleation site (NaX) in the gel, but also at the heterogeneous nucleation site (NaP) on the surface of the bentonite. In this case, due to the presence of more nucleation sites compared to the amount of gel supplied per each site, formation of NaP was preferred to occur instead of NaX zeolite.

The NaP/bentonite composite, which was prepared by surface modification, decreased the number of macropores compared to the original bentonite, resulting in a decrease in the specific surface area. However, due to the NaP zeolite's higher cation exchange capacity than bentonite, Cs and Sr removal efficiencies at groundwater conditions increased by 25% compared to the original bentonite.

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Keywords: Bentonite, Surface modification, NaP zeolite, NaX zeolite, Sorption efficiency

Development of a Canister Failure Module for the Adaptive Process-based Total System Performance Assessment Framework

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The APro (Adaptive Process-based total system performance assessment framework) is a COMSOL based 3D finite element analysis tool to simulate multiphysics in a geological repository system for radioactive wastes. Critical long-term processes potentially affecting safety functions of the repository can be analyzed by adapting a model with various modules included in the APro framework.

In this study, we developed canister failure module in the APro. The role of canister failure module is estimating the thickness of canister considering time-dependent corrosion rates. Currently, there are two available options in the module. The first option assumes that all corrosive chemicals reaching the canister surface immediately react with the canister. This option is appropriate for assessment scenarios where corrosion kinetics are limited by chemical transport. For scenarios where the kinetics are not limited by mass transport rate, on the other hand, application of the second option would be valid. In the second option corrosion rate is determined by multiple interfacial electrochemical and chemical reactions. Here, net current over the entire canister surface is set to be zero to satisfy electron balance (mixed potential theory).

For a flexible usability of the module, chemical species and kinetics models can be defined by users. As example cases, simulation results of the canister corrosion module are presented here incorporating electrochemical reactions by dissolved oxygen and sulfide. The presented 3D results showed that the corrosion rate is not homogeneous over the canister surface due to different oxidants concentrations over time. From the results, we expect that the developed module would be useful tool for investigating chemical stability of the canister considering the long-term evolution of the disposal system.

In the future, radionuclides release from the waste and complex geochemical reactions in a bentonite buffer will be added in or coupled with the canister corrosion module of APro.

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Keywords: Disposal canister, Corrosion, Numerical model, Finite element method

Development of Biosphere Assessment Module for Geological Disposal System

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A commonly considered method for the disposal of high-level radioactive waste (HLW) is the deep geological disposal, which isolates the waste deep underground. Geological disposal system is intended to isolate radionuclides in waste from the biosphere where living organisms inhabit over hundreds of thousands of years. However, radionuclides are possibly released from the repository and transported to the biosphere via groundwater flow if any hypothetical processes occur causing degradation and failure of safety barriers. Therefore, given this situation, the objective of biosphere assessment is to estimate the transport of radionuclides in the biosphere and determine the safety level of geological disposal system based on the risk and dose. For this purpose, this study has developed the biosphere assessment module to simulate the radionuclide transport through the biosphere, as a sub-module of APro (Adaptive process-based total system performance assessment framework). The biosphere assessment module has been developed using COMSOL for hydraulic and radionuclide transport processes. It simulates the pathway of radionuclides traveling from shallow aquifer to surface water body and soil. It was assumed that the transport of radionuclides was calculated in shallow aquifer and rivers only, while the other biosphere components were considered as recipients of radionuclides. In addition, the total time span of biosphere assessment is divided into time-independent snapshots based on the future development data. In each snapshot, the steady hydraulic condition is computed, and time-transient radionuclide transport is simulated by switching a hydraulic condition over snapshots. At the current stage, the applicability of the module was tested for a single snapshot period. The biosphere components and their properties were defined based on the ADioS (Artificial Disposal System). Plus, the radionuclide concentration on the GBI (Geosphere-Biosphere Interface) was assumed as a piecewise function. This study found that the radionuclide on the GBI was transported upward towards the water bodies which were groundwater discharge areas. The results suggested the potentially contaminated areas over the years. This module will be a useful tool to demonstrate the potential spatio-temporal variability of radionuclides in the biosphere under various scenarios and to provide the long-term radiological safety of the geological disposal system.

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Keywords: Biosphere assessment, Radionuclide transport, Surface water, Groundwater, Geological disposal system, COMSOL

Analytic Procedure of Rock Distribution by Drill Core for the Lithological Model of the Natural Barrier

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A methodology is under development to restore and predict the long-term evolution of the natural barrier comprise the site of radioactive waste disposal. For this purpose, surface geological outcrop, tunnel face and drill core are being used in research. Considering the condition that the radioactive waste repository should be located in the deep part, the drill core is an important subject that can identify deep geological properties that could not be confirmed near the surface. In this study, we established the process of selecting and analyzing deep drill core excavated around KURT to understand the deep geology of the study area.

First, classify the rock types of the drill core by appearance. It is divided into bedrocks and dykes with clear intrusion boundaries, and filled in 1 m sections. Granitoids, the bedrock of the KURT site, and the dykes were classified according to their grain size and mafic mineral content, respectively. Second, the results of the apparent classification are simplified to phase-1. Sections that appears thinner than 2.5 m, which is 5% of the sampling standard of 50 m, are excluded. However, in the case of the dikes that appear continuously within 2.5 m, they are left as a bundle. Third, sampling based on the phase-1 simplification. The sampling criteria are set at 50 m intervals, and additional sampling is performed when other lithofacies or altered sections appear within 50 m. Fourth, chemical analysis (XRD, XRF) is performed to analyze the mineral and main element content of the samples, and thin-section is prepared for microstructure analysis. Fifth, the results of the apparent classification and phase-1 simplification are combined and corrected with the data of geochemical and thin-section analysis. In the further study, the classification of granitoids categories will be supplemented through the results of geochemical and thin-section analysis, and finally input to the site lithological model.

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Keywords: Drill core, Rock distribution, Lithological model, Natural barrier

Feasibility Study of Surrogate Modeling and Coupling in the Process-based Total System Performance Assessment Framework

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Recently, KAERI has proposed a modularized framework of the process-based total system performance assessment, called APro, which simulates the radionuclide transport considering THC complex processes in a geological disposal system by coupling COMSOL Multiphysics and PHREEQC. Unlike the conventional system-level tools with simple and brief models, the process-based approach considers the long-term evolutions of the coupled processes accurately. However, the numerical simulation with the coupled processes is computationally challenging since it needs to deal with more than 10,000 boreholes and over 100,000 years of simulation time in the practical problems.

Thus, it is necessary to develop a new high-performance methodology which can effectively reduce the computational burden to utilize APro on a practical level. One good approach can be a surrogate modeling scheme using neural network, a popular branch of machine learning techniques in recent years. The surrogate model is supposed to rapidly emulate the original one while keeping a reasonable error. In this feasibility study, for example, a borehole domain having fine meshes is targeted as the surrogate model, coupled with the borehole boundary of remaining domain as a boundary condition. If the boundary quantities are properly responded by the surrogate model, it is expected to remove the numerous meshes inside the borehole and skip the burdening calculations for them.

This study aims to show the feasibility of surrogate modeling and coupling it with APro. Three-dimensional heat transfer problems in a geological disposal system with multiple boreholes are selected as test problems. To make full use of boundary condition interface of COMSOL Multiphysics in the coupling task, only boundary quantities of the borehole, temperatures and normal heat fluxes, are considered as input and output parameters to train the model. The current study successfully demonstrates the feasibility of the suggested method in the steady-state problems. The deployment is achieved by calling the surrogate model as a MATLAB function in the setting of boundary condition in COMSOL Multiphysics with the help of LiveLink for MATLAB. A shallow network is chosen as the network model for now, one of the simplest networks using fully-connected layers, but more advanced architectures will be studied further to demonstrate the scheme in the time-dependent problems.

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Keywords: Surrogate modeling, Machine learning, Neural network, APro, COMSOL multiphysics

Backfill Performance Targets and Their Deviating Loads, Conditions and Uncertainties

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Backfill is one of the important components of engineered barrier in an HLW repository, and is a barrier material that fills the deposition tunnel. The major safety functions of the backfill are to act as a hydraulic and mechanical barrier and thus to contribute to favorable and predictable hydrogeological, mechanical, and geochemical conditions for the buffer and canisters. The requirements of the backfill include: the backfill shall limit advective flow along the deposition tunnels; the backfill shall keep the buffer in place and contribute to prevent uplifting of the canister in the deposition hole; the backfill shall contribute to the mechanical stability of the deposition tunnels; the chemical composition of the backfill shall not jeopardized the performance of the buffer, canister or near-field bedrock. This study reviewed the performance targets for the backfill of advanced countries in HLW repositories, and analyzed the loads and their conditions and uncertainties in the conditions that may give deviations to the fulfillment of the performance targets.

The performance targets are the target values of the material properties that the backfill must have to fulfill its requirements, which are defined as measured properties by experiments or calculated properties by models. It has been reported that hydraulic conductivity, swelling pressure, structural stiffness, compressibility, and sorption equilibrium distribution coefficient are major performance targets of the backfill. The backfill performance targets, in general, are determined from laboratory and in-situ experiments at the expected disposal environment of the site and repository. However, a literature review has shown that deviations from the fulfillment of backfill performance targets may arise primarily from considerations of evolution and interactions with other components. The time-dependent and space-dependent loads of the evolution and interactions have involved: piping and erosion by groundwater inflow to deposition tunnels, backfill alteration such as illitization and cementitious silicification, backfill geochemistry strongly influenced by that of the surrounding plug and host rock, homogenization in the backfill and at backfill-pellet-buffer-rock interfaces, chemical erosion due to dilute groundwater conditions during the ice-melting period after the first glacial cycle, complex thermal-hydro-mechanical-chemical (THMC) processes in the backfill. The deviations from the fulfillment of backfill performance targets were also sensitive to the conditions for the considered evolution and interactions and uncertainties in the conditions. The result of this study will be used as basic data for the development of performance targets for backfill to be applied to a high-level waste repository in Korea.

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Keywords: Backfill, Performance targets, Technical requirement, Deviations from the fulfillment of backfill performance targets

Analysis of Long-term Evolution Phenomena Applied to the Safety Assessment of Major Disposal Countries

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Since it takes hundreds of thousands of years for the radiotoxicity of spent nuclear fuel to decrease to natural levels, the safety assessment of the repository is performed over a period of more than one million years. In this long period, interactions between each repository barrier and climate change are inevitable. The long-term evolution phenomena applied to the safety assessment of each country are analyzed to develop FEPs and scenarios.

The long-term evolution of the repository can generally be divided into three stages: the initial stage after operation and closure, and before/after rapid climate change. The THMCR interactions between the internal components of the repository are the principal phenomena in the stage before climate change. The ice age is the capital phenomenon considered in the stage for very long-term evolution. The ice age developments e.g. thickness of ice sheet and formation of permafrost can be divided into three groups according to latitude. The land uplift/depression due to glacial formation and surface erosion are considered together.

The FEPs and Scenarios for long-term evolution will be developed to include most of the major long-term evolution phenomena defined in this study and which are highly probable in domestic disposal conditions. Also, the database for the evaluation of long-term evolution scenarios will be constructed in the future.

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Keywords: Repository for spent nuclear fuel, Long-term evolution, Performance and safety assessment, THMCR, Climate change

Groundwork for the Establishment of APro in Massively Parallel Computing Environment

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Increased computing power and expansion of cloud computing services have made it possible to simulate large-scale complex multi-physics phenomena. The growth of processing power has been driven by an increase in clock speed and an increase in the number of cores which results in the parallel computing. And the parallel computing environment can be easily constructed with the help of the growth of cloud computing services, such as Google Cloud, AWS, Azure and Oracle, which provide on-demand computing infrastructure platforms.

In order to properly utilize the APro code, **Adaptive Process**-based total system performance assessment framework for a geological disposal system, for practical problems, a parallel computing is essential to address heavy computational burden. The APro modeling interface is constructed with MATLAB combining COMSOL and PHREEQC for the simulation of coupled multi-physics phenomena with geochemical reactions. The model complexity coming from the coupled multi-physics can be handled by using appropriate numerical approaches. Application of APro for the large-scale problem, for example, a disposal system having thousands of boreholes, requires large memory so that the parallel computing environment with extended resources is necessary.

This study performs the groundwork for the establishment of APro in massively parallel computing environment. Proper computing resources in the cloud computing services are selected for running APro with the benchmark problem considering a THC coupled processes. In order to minimize the internode communication, the appropriate number of boreholes per single node is estimated to utilize the most of memory resources for each node. To estimate the appropriate number of boreholes for several candidates of the numerical approaches, memory requirements are compared as the number of boreholes increased and the most suitable numerical approach is chosen according to the parallel computing efficiency and memory requirement. In addition, it is examined that APro calculation is successfully performed with the multi-node parallel computing environment so that the geological disposal system with thousands of boreholes can be handled.

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Keywords: APro, Total system performance assessment, COMSOL, Massively parallel computing, Cloud computing service

A Comparative and Parametric Study of Fracture Modeling Approaches for Nuclide Transport in Crystalline Rock

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Crystalline rock has been considered as a host rock for the high-level radioactive waste repository due to its advantages such as stability at high temperature, reasonably high thermal conductivity and very low permeability in the absence of fractures. However, most crystalline rock masses contain fracture systems, which have much higher permeability than rock mass. Since fractures with high permeability has been widely known as the main path for fluid flow and nuclide transport, it is very important to reflect fracture network in subsurface precisely for numerical studies. For more precise estimation of nuclide transport behavior in long term duration, different kinds of models that reproduce fractures in rock matrix have been suggested.

There are two most widely used fracture modeling approaches: (1) Discrete fracture network (DFN) and (2) equivalent continuum porous medium (ECPM). While DFN characterizes individual fractures explicitly with fracture properties such as shape and size, ECPM simulates fractures implicitly by averaging hydraulic properties of fractures into those of equivalent porous media. However, each approach has problems related to geometric complexity and accuracy loss by averaging respectively.

In this study, a benchmark problem of DECOVALEX-2023 Task F, which is the international research and model comparison collaboration, is simulated using Discrete fracture matrix (DFM), which express deterministic fractures explicitly in domain and minor stochastic fractures implicitly with upscaled quantities, to minimize the accuracy loss from two existing models. The transport of conservative tracer was evaluated through breakthrough curve (BTC). The characteristics of three different models (DFN, ECPM, DFM) are evaluated by comparing the results of each case. Thereafter, the parametric study was also performed to explore the effect of different parameters on nuclide transport. This study is expected to provide benefits in developing reliable performance assessment model in the future.

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Keywords: Discrete fracture network (DFN), Equivalent continuum porous media (ECPM), Discrete fracture matrix (DFM), DECOVALEX-2023 task F, COMSOL multiphysics

The Preliminary Study on the Fracture Properties and Hydrogeological Characteristics in the Granitic Rock at 1 km Depth

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While the geological repository has considered the final management methodology for high-level radioactive waste (HLW), some leading countries in the field of HLW disposal research, including the United States, is proposing deep borehole disposal (DBD) to supplement the geological repository, or suggest an alternative to the deep geological repository. Until now, conceptual studies have been conducted based on the fundamental and standard design of DBD in Korea. Furthermore, it is difficult to evaluate the applicability of DBD in Korea due to the lack of site characteristics in the deep areas (below the 3 km from the surface). Research on the deep environment has been attempted in some industries, such as geothermal energy use and CO₂ storage underground, but no site characterization was conducted depending on the depth required for safety evaluation of DBD because of the limited depth. It was not until recently that the Korea Atomic Energy Research Institute (KAERI) planned to drill the deep borehole for understanding the site characteristics around the depth of DBD. The borehole will be drilled up to 2.5 km deep into the crystalline rock by 2025 and after that, expand the depth to 3 km below the surface in the study area.

This study describes the fracture properties and hydrogeological characteristics evaluated in the borehole at a depth of 1 km drilled in the study area as a preliminary study for evaluating the applicability of DBD in Korea. The frequency of fracture in the borehole tends to decrease with depth, especially in areas near the surface, 500 m deep, and 800 m deep. Fracture zones consisting of clusters of fractures exist up to 500 m underground, but little fracture zones are expected to exist in some sections at depths below 500 m. Meanwhile, in the hydraulic test conducted by setting a test section of 100 m, the permeability of the rock was high due to the influence of the fracture zone. However, the hydraulic conductivity of the rock without fractures can be found to be low permeable, with a value of $1 \times 10^{-11} \text{ m} \cdot \text{sec}^{-1}$ or less, and the hydraulic conductivity decreases with depth.

Keywords: Geological repository, High-level radioactive waste, Deep borehole disposal, Site characterization

Microstructural Analysis of Host-rocks and Altered-rocks for Identifying the Rock Distribution Around KURT Area

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Disposal facilities for radioactive waste shall be sited to provide isolation from the accessible biosphere. The features shall aim to provide this isolation for tens of thousands to a million years after closure. For the safety assessment of the repository, the long-term natural evolution and possible events of the site, that can cause disturbances to the facility over the period of interest, should be considered. Studying the geological development processes that the site has experienced can contribute to understanding and describing the present-day conditions. Moreover, knowledge of the past is necessary to predict the future evolution of the site. To describe this long-term geological evolution, information from the site investigation is needed. The database for analyzing geological development processes can be represented as an integrated model, covering the current state of the geological environment and natural processes that affect long-term safety. Lithological model is the most basic component of the integrated model. The area around KURT, which is an underground facility located in a massive Jurassic igneous body, have been investigated as a test site of disposal research. In this research, we studied the differences in petrological and microstructural characteristics of intact host-rocks, dykes and altered rocks, to construct the methodology for the development of lithological model of a disposal site. Rock-cores drilled around the KURT were used to understand the lithological distribution deep underground. The results can also contribute to the construction of hydraulic-mechanical model by identifying the differences in physical properties of different rock types.

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Keywords: Radioactive waste disposal, Site descriptions, Long-term safety, Lithological model, Microstructures

Development of Radionuclide Transport Model in the Near Field Considering Non-isothermal Two-phase Flow Using COMSOL Multiphysics

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In recent years, due to the demand for the process-based performance assessment of a geological radioactive waste disposal, APro (Adaptive Process-based total system performance assessment framework for a geological disposal system) was developed by Korea Atomic Energy Research Institute (KAERI). One important issue is the leakage of radionuclides from the canister in the early stage of disposal, where the bentonite buffer is partially saturated and heat is generated from the canister.

In order to demonstrate the transport of radionuclides in this situation, two-phase fluid flow should be considered because the bentonite buffer is partially saturated which means the pore space is filled with water and air. In addition, the temperature increases due to the heat generated from the canister and the mechanical behavior of the medium also affect the fluid flow as well as the migration of radionuclides, so these processes should be considered. Therefore, to solve this problem, analysis of coupled thermal-hydraulic-mechanical (THM) processes with non-isothermal two-phase flow should be accompanied, and at the same time, transport of radionuclides also needs to be linked with these processes.

In this study, the transport module of radionuclides with considering non-isothermal two-phase flow was developed using COMSOL Multiphysics for the near field as a part of APro. The equation of mass conservation was used as the governing equation with the effects of decay and sorption for the radionuclides transport module, and this module was implemented to COMSOL Multiphysics using the "Transport of Diluted Species in Porous Media" physics. Using the developed numerical module, the applicability was evaluated through the analysis of radionuclide migration for a single disposal hole.

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Keywords: Geological radioactive waste disposal, Process-based total system performance assessment, APro, Radionuclide transport in a porous media, Non-isothermal two-phase flow

Mechanisms of Gas Generation and Scenarios of Gas Migration in Korean SNF Repository Site

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Study for the effect of gas on the radionuclide migration in the SNF (spent nuclear fuel) repository barriers becomes necessary to mitigate the global concerns such as radiation leak and radioactive contamination around the SNF repository sites. Mechanisms of H_2 gas generation due to the corrosion of copper SNF disposal canister and to the biochemical reaction in barrier were reviewed from previous researches. The H_2 gas generation originated from the canister corrosion in redox condition of groundwater was investigated and its migration in the artificial barrier was simulated by the laboratory scale experiments. The compacted bentonite block (dry density: $1.6 \text{ g}\cdot\text{cm}^{-3}$) was manufactured and its physico-chemical properties were identified to evaluate its feasibility as the artificial barrier surrounding the SNF canister. The H_2 gas injection pressure into the compacted bentonite core as an artificial barrier was investigated on the simulated repository site conditions in experiments. Quantification of gas movement in pore space through micro-scale visual experiments were also performed. The plausible scenarios of gas migration in the artificial barrier surrounding the SNF disposal canister were also developed and the gas migration into the barrier medium as a visco-capillary flow and a dilatant flow were also investigated in laboratory experiments. The cylindrical core (5 cm in diameter and 5–10 cm in length) from the compacted bentonite block was used in experiment. The high pressurized stainless cell with the facility controlling the confining pressure, piston pump, and back pressure regulator were used. The gas entry pressures with various confining pressures and water saturation ratios for the core were measured and the gas breakthrough curves in the core were also acquired to calculate the effective gas permeability and the gas flow velocity of the core in the specific confining pressure and saturation condition.

From experimental results, the H_2 gas entry pressure on various confining pressure and water saturation conditions and the change of its flow rate in the compacted bentonite core were determined. The physical and mineralogical changes of the compacted bentonite core after the experiment were also measured to evaluate its resistance against the chemical reaction. Information and data from this study will be applied to determine the most suitable barrier medium for the Korean SNF repository site and to support meaningful parameter values for the long-term gas migration modeling around the repository site.

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Keywords: Artificial barrier, Bentonite, Dilatant flow, Gas migration, Repository site, Spent nuclear fuel

Verification of APro Through a PHREEQC Benchmark Problem

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APro, developed by KAERI, is a process-based total system performance assessment model for a geological disposal system. APro can simulate the radionuclide transport affected by thermal, hydraulic, mechanical and geochemical changes that may occur in the engineering and natural barriers of the disposal system. For this, APro adopts the sequential non-iterative approach (SNIA), one of the operator splitting (OS) methods, to combine COMSOL and PHREEQC to consider complex geochemical reactions and multi-physics together. Several codes, including iCP from Amphos 21 and IPSACC from the University of Osaka have used same OS method to combine COMSOL and PHREEQC, and verified the accuracy of the developed code by comparing with other OS-based codes through a simple reactive transport problem. Since APro also needs to verify its accuracy in using SNIA, it is necessary to solve simple benchmark problems covered by other codes and compare the results before applying it to the safety assessment for the disposal system.

In this study, a benchmark simulation for “Example 11 – Transport and Cation Exchange” problem, one of the representative examples of 1D reactive transport in PHREEQC, is conducted within the APro framework and the results are compared with those from PHREEQC. The Example 11 is a problem that simulates the chemical composition of the effluent when the calcium chloride (CaCl_2) solution flows into a column containing sodium-potassium-nitrate (Na-K-NO_3^{2-}) solution that is in equilibrium with the cation exchanger. In this case, as entered Ca forms a new equilibrium with existing K and Na, the chemical composition of the effluent changes, and also the advection and dispersive mixing give a great influence on the changes in the chemical composition. Within the APro framework, the chemical compositions of the effluent are obtained by simulating the non-reactive transport of the solutes in COMSOL and the equilibrium reaction of the solutes with exchanger in PHREEQC. As a result, it is shown that there is no significant difference between results from APro and PHREEQC.

In the future, the accuracy of APro will be proven through other benchmark problems, and optimal spatial and temporal discretization will be suggested through the OS error analysis.

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Keywords: APro, Total system performance assessment model, Operator splitting, Sequential non-iterative approach, Model verification

4분과

중저준위폐기물관리 (Oral)



A Study on the Revision of the Current WCP-related Regulations to Systematically Implement WCP

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Domestic WCP (Waste Certification Program)-related regulations stipulate that generators can establish and operate WCP according to the notice of the Ministry of Trade, Industry and Energy. In accordance with the recent amendments to the Waste Management Act, the Nuclear Safety Commission requires the generators to provide a quality assurance plan for waste management.

In order to complement the limitations of these regulations, it is necessary to clarify the role and scope of the administrative disposition (suspension of approval, revocation of approval, re-approval after revocation of approval) and the role and scope of the regulatory body when operating the WCP of the generators.

Accordingly, this study reflects the results of analysis of domestic and foreign WCP-related regulations, secures overall interconnection between generators, management agency and regulatory body related to WCP operation, and set considerations for revising WCP-related regulations (proposal).

Based on the considerations to be prepared for the systematic operation of the WCP, a revision (proposal) of the WCP-related regulations was presented through an impact analysis.

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Keywords: WCP, WCP-related regulations, Generators, Management agency, Regulatory body

Status of IAEA Safety Standards for Clearance

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Since Safety Series No. 89 (1988), which provides the concept of trivial risk and dose, the IAEA has published safety standards, guidances and technical reports for clearance such as SS-115 (1996), RS-G-1.7 (2004), Safety Report No. 44 (2005), SR-67 (2012) and the Basic Safety Standards (GSR Part 3, 2012). In particular, the dose criteria for clearance was changed in GSR Part 3 to $10 \mu\text{Sv}\cdot\text{a}^{-1}$ for realistic scenario and $1 \text{ mSv}\cdot\text{a}^{-1}$ for low probability scenarios.

Recently, the IAEA is in the process of revising RS-G-1.7 and is preparing to publish a Draft Safety Report on the derivation of specific clearance levels (CLs) in materials being suitable recycling, reuse, or for disposal in landfills.

In this study, the status of IAEA safety standards for clearance and related major activities were reviewed.

There are some differences in terminology and approach between SS-115 and GSR Part 3 that necessitate revision of RS-G-1.7 published prior to the publication of GSR Part 3. Specifically, the requirements in SS-115 apply to practices and interventions while GSR Part 3 is structured around three different types of exposure situations (planned, emergency, and existing). The concept of clearance in planned exposure situations is included in GSR Part 3, but supporting guidance has not yet been developed. Therefore, a new Safety Guide for the Application of the Concept of Clearance (DS500) was proposed. DS500 will provide more information on clearance process, responsibilities of the licensee and the regulatory body and technical and safety implications of clearance. Especially, technical issues discussed in DS500 are clearance options, final monitoring for demonstrating compliance with CLs, sampling, uncertainties, mass-specific ($\text{Bq}\cdot\text{g}^{-1}$) and surface specific ($\text{Bq}\cdot\text{cm}^{-2}$) CLs and measures and etc.

While CLs in RS-G-1.7 assume the use of the cleared material without any further restrictions, decommissioning waste may be generated in large quantities containing low levels of activity and there are demands for a simple methodology to evaluate disposal options for these wastes. Accordingly, the IAEA derived specific CLs for disposal of waste in municipal landfills and is preparing a new safety report entitled as “Derivation of specific clearance levels in materials being suitable recycling, reuse, or for disposal in landfills”. In the draft safety report, SR-44 scenarios were adjusted, and scenarios for water-pathway was added and re-evaluated. In addition, the landfill-disposal model was reviewed and updated. Also, the new report will provide the methodology for the derivation of specific CLs considering status of landfills, exposure pathway, period of institutional control for different types of landfills.

Clearance is an important option for management of material and waste from operation and decommissioning of facilities, it enables significant reduction of amounts to be managed as radioactive waste. The IAEA initiated the revision of existing guidance on clearance and several supporting publications and it is expected to continue to promote the maintenance of the safety document system for clearance.

Keywords: Clearance, Specific clearance, Landfill disposal, Safety standards

Implementation of Diffusional Release Source Term Model Using GoldSim

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One of efforts to be made for enhancing safety margin by realistic safety assessment of the silo-type Wolsong LILW disposal facility is to consider the effects of diffusion-release waste forms in the source term modeling. In solidified waste forms, radionuclides are released into the concrete disposal container within the silo through diffusion to the surface of the waste forms.

In this study, a diffusional release source term model was implemented in GoldSim modeling. Each of the waste types in a silo was represented in the modeling as a cylinder. Using the GoldSim cell pathway approach, a diffusion-release waste form is divided into a series of concentric cells in a cylinder. Through testing of degree of discretization, five concentric cells are used in the modeling. The fluxes between cells were computed using diffusive mass flux links in GoldSim. The diffusive mass flux [mass/time] of radionuclides between two neighboring cells was calculated in accordance with Fick's First Law as the diffusive conductance [volume/time] and the concentration difference [mass/volume] between the cells. The diffusive conductance was represented as the function of diffusive area, diffusive lengths and effective diffusion coefficients in adjacent cells. The diffusive area is the area of the cylindrical boundary between the concentric cylinders, and diffusive length is the thickness of the walls of the five modeled concentric cells for each cylinder. Near-field mass flux to the geosphere and far-field mass flux to biosphere for normal groundwater release scenario were compared with those of non-diffusional release models in the previous GoldSim modeling.

It was found that, in general, diffusional release model tends to give later peak times and lower peak fluxes than non-diffusional release model. For radionuclides having short half-lives (H-3, Co-60, Ni-63, Sr-90, Cs-137) it showed that a significant difference in timing and magnitude of peak flux results. This is due to large mass reduction in the barrier during transport through diffusion phenomena in these radionuclides. In contrast, there was no significant differences in both models for radionuclides with low distribution coefficients and long half-lives (C-14, Tc-99, I-129, Ni-59).

When radionuclide-specific effective diffusion coefficient on cement solidified wastes is obtained, it is expected that this model could be applied in the realistic source-term modeling for the development of safety case for the Wolsong LILW disposal facility.

Keywords: GoldSim, Safety assessment, Diffusional release model, Disposal facilities

Deep Learning-based Object Classification of Radioactive Waste From Research Reactor

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In this presentation, we will introduce the data preprocessing process, the training method of the deep learning model, the performance evaluation of the deep learning model, and the application of explainable AI for the classification of radioactive waste images.

The classification of Radioactive waste by raw materials is the correct way to reduce waste and minimize danger. For example, waste containing radionuclides with compressible waste may be separated from non-compressible waste, or flammable waste may be separated from non-flammable waste. The efficiency and quality of the classification are highly dependent on the purity and accuracy of the sorted raw material. However, the classification process usually required a lot of labor and is subject to human errors.

Deep learning techniques help in the automatic classification of waste types accurately and rapidly. Since the lack of enough waste images, we extract image and labeling information (waste type, date of creation, etc.) from the document file by applying the robotic automation process (RPA) using python library and win32 API. In order to overcome data shortage and improve performance, we applied various augmentation methods like cut-mix and pre-trained deep learning models during the training process. The accuracy of the deep learning model on the test set was comparatively accurate when considering the data size, and one of the explainable AI techniques, Grad-CAM, was introduced to confirm the reliability of the results. If more extra data is collected and the performance of the deep learning model is improved, it is expected that human errors that occur in the radioactive waste disposal process will be dramatically reduced.

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Keywords: Radioactive waste, Research reactor, Image classification, Grad-cam, Explainable AI, Robotic process automation

Non-destructive System and its Calibration Method for Isotope Identification and Quantification of Radio-waste From Decommissioning

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Radio-wastes generated from decommissioning process include large amount of contaminated concrete, soil, metals and miscellaneous materials. Unlike the standard packaging methods using 200 and 320 liter drums, the waste from commissioning process requires larger size containers for practical, economical and effectiveness point of views.

We have been working together with KHNP and KORAD to develop a non-destructive system to measure activity concentration of isotopes in the wastes from decommissioning packaged in diverse size of containers.

The system consists of total 2 columns of detector modules and each column has 2 HPGe detectors. Each HPGe detectors are allowed to move up/down within the column. And each column can move in/out toward the waste container. This detector movement allows to accommodate different size of waste packaging containers. In addition, this enables adjust the UFOV (useful field of view) of the system. The container is positioned on a linear track and moves longitudinal direction for scanning while the detector column is stationary position.

Especially, in this paper, we present methods to calibrate the system developed. Calibration includes, 1. Method to determine UFOV, 2. Calibration volume and 3. Method to estimate efficiency curve.

Based on our methods, the optimum detector collimation was selected 140 degrees and the segmentation was determined up to 24 volumes for $1 \times 1 \times 3 \text{ m}^3$ container case. The efficiency curves as a function of gamma energy and material density were estimated based on combination of Monte-Carlo simulation and experiments.

The proposed method shows about 2 times better MDA (minimum detectable activity) performance compared to that of commercially available systems by Mirion-Canberra and Ortec.

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Keywords: Decommissioning, Activity concentration, Isotope identification, Radio waste, Calibration

Radioactive Cs Immobilization by Using Cs Pollucite Ceramic Wasteform

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Radioisotopes of cesium presented in spent nuclear fuels are ¹³⁵Cs ($T_{1/2} = 2.3$ million years) with long-live nuclide and ¹³⁷Cs ($T_{1/2} = 30.2$ years) with high heat generating nuclide. Because of the heat generation property of radioactive cesium, it is difficult to apply immobilization methods using a general glass wasteform matrix such as borosilicate glass with high Cs waste loading. Generally, ceramic hosts have higher density and chemical durability than glass hosts and can incorporate complex chemical species in the lattice structure. Various ceramic wasteforms for Cs immobilization have been researched such as aluminosilicate ($\text{CsAlSi}_2\text{O}_6$), phosphate ($\text{CsZr}_2(\text{PO}_4)_3$), titanate ($\text{Cs}_x\text{Al}_x\text{Ti}_{8-x}\text{O}_{16}$, $\text{Cs}_2\text{TiNb}_6\text{O}_{18}$) and $\text{CsZr}_{0.5}\text{W}_{1.5}\text{O}_6$. Cs pollucite is Cs-bearing zeolite with high thermal stability and low solubility. Due to its high structural stability, the captured cesium is not released unless the framework is destroyed.

In this study, Cs pollucite powder was produced using kaolinite ($\text{Al}_2\text{Si}_2\text{O}_5(\text{OH})_4$) and $\text{Cs}_2\text{CO}_3(\text{s})$ as precursor as a result of a gas-solid reaction. Kaolinite is decomposed into mullite and silicate during heat treatment at $1,200^\circ\text{C}$. Mullite is composed of silicate and alumina to form CsAlSiO_4 with gaseous Cs_2O . Cs_2CO_3 is decomposed to Cs_2O and CO_2 at 610°C . CsAlSiO_4 is converted into Cs pollucite ($\text{CsAlSi}_2\text{O}_6$) by sintering at $1,200^\circ\text{C}$ for 2 hours. Cs pollucite powder was prepared different Cs-loading by adjusting the mole ratio of $\text{Cs}_2\text{O}/(\text{mullite}+\text{silicate})$ to 0.5, 0.8, 1.0, 1.2, 1.4 and 2.0. Subsequently, XRD, FE-SEM and SEM-EDS were employed to analyze the crystal structures, morphology and elemental distribution, respectively. PCT-A test (ASTM C1285) was performed at 90°C for 7 days. Leachates were analyzed to confirm the leaching rate of Cs by ICP-MS. and Si, Al composing matrix components by ICP-OES. Optimized ratio with highest Cs pollucite purity was $\text{Cs}_2\text{O}/(\text{mullite}+\text{silicate})$ mole ratio = 1.2, and pellets were produced using that. The Cs pollucite powder was mixed homogeneously with 1.5, 2.0wt% polyvinyl alcohol as a binder and dried at 70°C for overnight to make powder for pellet preparation. Thereafter, 8 g powder was pressed using the 25 mm punch-die apparatus at 50, 100, 150, 200, 400, 600 bar for 1 hour, and then sintered at $1,400^\circ\text{C}$ for 5 hours to produce a monolithic pellet with a size of dia 25 mm×10 mm. To evaluate applicability of the wasteform prepared, physical properties of the pellets were performed to analyze a compression strength, thermal conductivity and thermal expansion.

Keywords: Radioactive cesium, Cesium pollucite, Ceramic wasteform, Sintering method, Palletization

A Case Analysis of Overseas Disposal Facility for Derivation of Organic Complex Agent Acceptance Criteria

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In a radioactive waste disposal facility, the presence of complex compounds may decrease the adsorption capacity of radionuclides and increase the mobility of radioactive waste. Therefore, in order to regulate the concentration of complex agents that combine with radionuclides to form a complex compound, each country specifies the acceptance criteria. In the case of Korea, the disposal facility operator confirms the suitability of the disposal of the radioactive waste requested by the generator in accordance with the acceptance criteria, and there are standards and inspection methods for chelating agents. However, there is no acceptance criteria for wastes containing cellulose that are decomposed to form ISA, another representative organic complex agent. Therefore, through this case analysis of the acceptance criteria for overseas disposal facilities, we will try to find a method necessary for deriving the acceptance criteria for materials containing organic complex agents in Korea.

The US NRC standard specifies that 1% or more of all chelating agents must be reported. Some states limit chelating agents to less than 8%, while others require waste stabilization. The acceptance criteria for organic complexes at the Drigg disposal facility in the UK are limited to chelates, and if they contain more than 0.1%, it is impossible to dispose of them. In particular, through an experiment on ISA that affects the behavior of nuclides in a cellulose dispersion, it was found that there was no effect on the increase in the solubility of nuclides, so no separate cellulose-related criteria were specified in the acceptance criteria. Sweden is actively conducting research on cellulose as well as chelates. Belgium specifically specifies and limits the amount of cellulose present in radioactive waste.

Through these case analysis of overseas organic complex agent acceptance criteria, a direction was sought to prepare Korean criteria for organic complex agent, a substance that accelerates the movement of radionuclides in relation to the long-term safety aspect of the repository.

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Keywords: Complex agents, Acceptance criteria, Organic complex compound, Long-term safety

Structural Stability Assessment of 16-pack Disposal Container in the Silo-type LILW Repository

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The construction and operation of radioactive waste repository are essential to dispose of wastes safely in the countries operating nuclear power plants. Therefore, the radioactive waste repository should maintain its structural stability for a long enough time until the radiological risk of waste decreases sufficiently. Among the structures of repository, the disposal container should be intact for a long enough period as it directly contains radioactive wastes. One of the main factors that affect to the structural stability of the disposal container is static loading contributed by the weight of backfilling materials. Thus, the effect of weight of backfilling materials to the structural stability of the container should be examined.

In Korea, low- and intermediate level waste (LILW) repository is in operation which is silo-type repository located below ~100 m depth. In the silo, waste packages consist of disposal container and radioactive waste are stacked vertically. After closure of the repository, empty spaces are backfilled with backfilling material which is mixture of crushed granite and shotcrete which applies ~0.25 MPa to the waste packages at the top position. From the modeling result in previous study, the plastic behavior of the 16-pack type disposal container is checked. As the plastic strain is directly connected to the fracture for concrete material, the occurrence of plastic strain is severe problem. In addition, previous modeling results showed that the disposal container endures the expected loading by adding additional supporting pole at the empty spaces between wastes. However, the verification of the modeling result should be conducted to verify results of modeling by conducting experiment.

In this study, fracture experiments of mock 16-pack disposal container are conducted. The mock 16-pack disposal container is 1/6 scaled size of its original dimension. To describe the loading by the weight of backfilling materials, the steel bearing balls are stacked above the mock disposal container. The lead blocks are additionally stacked above the bearing balls by adding steel loading plate between ball and block. The pressure applied to the container is divided into two cases: maximum ~30 kPa for empty disposal container and 0.5 MPa by using compression machine for reinforced container. By installing strain gauges on the lid, supporting poles and wall of disposal container, the strain changes are aimed to be checked. As a result of this study, the strain behavior and fracture loading are compared with the modeling results.

Acknowledgements

This work was supported by the Nuclear safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 1903005).

Keywords: Structural stability, Repository, Disposal container, Silo-type, Backfilling material

4분과

중저준위폐기물관리 (Poster)



A Hybrid Advance Oxidation Process for Effective Treatment of Spent Resin

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Ion exchange resins (IERS) have been widely deployed in nuclear power plants (NPPs) for the removal of cationic / anionic radionuclides (e.g., ^{235}U , ^{238}U , ^{60}Co , ^{90}Sr , ^{137}Cs , ^{65}Zn , ^{51}Cr , ^{58}Ni , ^{59}Ni , $\text{D}^{14}\text{CO}_3^-$ and $^{14}\text{CO}_3^{2-}$) from primary circuit and other systems. During operation of NPPs, these IERS are regularly replaced to maintain the highest possible water quality and operation efficiency, because they start losing their effective properties due to physical, chemical, and radioactive deteriorations. Therefore, International Atomic Energy Agency (IAEA) had enlisted the “spent resins” as hazardous radioactive solid waste, and it is highly recommended that these wastes should be properly treated and disposed to reduce their hazards to the human and environment. Among various treatments, advance oxidation process (AOP) has been an attractive choice in terms of the higher volume/weight reduction of spent resin wastes. In this study, we performed a hybrid treatment Fenton- Like dissolution followed by sulfate-radical advanced oxidation processes SR-AOPs for the effective mineralization of nuclear grade cationic IRN-77 resin. The optimal reaction parameters for SR-AOPs of dissolved IRN-77 resin were determined, and the initial pH, PMS dosage, and reaction temperature were found to be the most influential parameters for COD removal (%) of resin. The experimental results demonstrated that 90% COD removal ($1,000 \text{ mg}\cdot\text{L}^{-1}$) of dissolved IRN-77 can be achieved for SR-AOPs by optimizing the reaction parameters as initial pH=9, $\text{Co}^{2+}=4 \text{ mM}$ (catalyst), PMS = 60 mM (as oxidant) at 60°C temperature for 60 min reaction. The electron spin resonance spectroscopy (ESR) spectra confirmed the presences of SO_4^- and OH^- radicals. In addition, non-targeted GC-MS and Fourier transform infrared spectroscopy (FT-IR) analyses were performed for the identification of organic compounds and their functional groups in dissolved IRN-77 resin. Based on the current results, we believe that use of Co^{2+} /PMS based AOPs can be a green alternative for mineralization of spent resin with waste volume reduction.

Acknowledgements

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Keywords: IRN-77 resin, Advance oxidation process Cobalt, Peroxymonosulfate, Sulfate radical

Analysis on Ultrasonic Attenuation of Concrete Depending on the Temperature, Dry, and Saturation Conditions

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The purpose of this study is to determine the ultrasonic attenuation coefficient (α) of the concrete in the low-intermediate level nuclear disposal environment. To conduct the laboratory tests, The samples were exposed to the temperature (15°C, 45°C, 75°C), dry, and saturation conditions for 32 days after curing for 28 days. The attenuated ultrasonic signals according to the transmission distance were measured to determine α s. The results showed that AE absolute energy in the saturated concrete was higher than that of the dry concrete in the initial range of the signal transmission distance, but the α of the saturated concrete was higher rather than that of the dry concrete. This seems due to the fluid viscosity effect that occurs in concrete micro-cracks as the signals pass through the concrete medium and various fluid flow mechanisms induced by signal transmission. As regarding the temperature conditions, the α tends to decrease as the temperature increases regardless of the saturation and dry conditions. This result seems that the higher the temperature of the concrete, the lower the water content that seems to be related to the water evaporation effects. However, the α was more sensitive to the saturation and dry conditions than the temperature conditions, which means that the saturation and dry conditions is the main consideration when measuring the AE attenuation of a concrete. The determined α expect to be used to evaluate the integrity of Gyeongju silo concrete in a low-intermediate level nuclear disposal environment by estimating the actual AE absolute energy at the location of cracks, and to determine the optimum placement of AE sensors.

Acknowledgements

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Keywords: Attenuation coefficient, Ultrasonic, Concrete, Low-intermediate level nuclear disposal environment

A Study on a Purification Process of Sr Nuclide From Salt Waste Generated From Pyrochemical Process

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Sr-90 nuclide produced by nuclear fission is beta emitting radionuclide. It decays sequentially into Y-90 and Zr-90 through beta decay and the heat released during beta decay could be used as a source of radioisotope thermoelectric generator (RTG). Also, Sr-90 and Y-90 are used as a source of beta particles and extensively utilized in medical and industry fields. In particular, Y-90 has high beta radiation energy and short half-life, so it is used for radiation therapy. Therefore, high-purity Sr nuclides are required for the application of Sr-90 and Y-90 to increase the thermal efficiency of nuclear battery operation or to prevent side effects on the human body during its use in medical therapy. Sr-90 is present in a spent nuclear fuel and can be separated via reactive-distillation process during pyrochemical process, a spent nuclear fuel recycling process developed by the Korea Atomic Energy Research Institute (KAERI), where Ba nuclide is obtained together by co-precipitation due to chemical similarity. In addition, Sr-90 finally decays into Zr-90 during the storage period after separation. Therefore, Ba and Zr should be separated to obtain high purity Sr nuclides. In this work, the purification process using chromate and carbonate precipitation reaction was developed for recovering purified Sr nuclide from a waste mixture containing Sr, Ba, and Zr. A feasibility and possibility of purification process were confirmed through the preliminary study and aqueous chemistry simulations using MINTEQ 3.1 software. Based on these results, modified purification process via pH control was established. A systematic study was conducted to optimize Sr purification processes, and as a result, co-precipitation of Sr could be effectively reduced in the precipitation reaction for removal of impurities. The purified final product (SrCO_3) showed a high yield exceeding 84.3% and a high purity exceeding 99.9%.

Keywords: Strontium-90, Nuclide separation, Purification, Precipitation, Spent nuclear fuel recycling, Pyrochemical process

Substantiation and Verification Using Nanoparticle Complex-fluid Decontamination Device in Nuclear Power Plant

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This study investigated the verification and applicability of the ‘Nanoparticle Complex-fluid Decontamination Device’ that decontaminates radioactive metal wastes generated in large quantities when operation/decommissioning of the nuclear power plants with nanoparticle complex-fluid decontamination technology. The nanoparticle complex-fluid decontamination device is linked to decontamination technology and liquid waste treatment technology. It decontaminates the surface of metal radioactive waste using an oxidative decontamination foam, and can reduce secondary waste generation about 1/10 compared to wet decontamination technology. Radioactive metal wastes generated from nuclear power plants were secured to perform the technical verification and applicability evaluation of advancement equipment. As a result of radionuclides analysis, Co-58 and Co-60 ($4.64 \times 10^{-2} \text{ Bq} \cdot \text{g}^{-1} - 3.97 \times 10^0 \text{ Bq} \cdot \text{g}^{-1}$) were detected at the VLL (Very Low Level). Although metal waste with VLL contamination as a result of the decontamination using an advanced equipment was secured the 100 decontamination factor (DF) and was found to be below the MDA (average MDA: $1.17 \times 10^{-2} \text{ Bq} \cdot \text{g}^{-1}$), which satisfies the allowable concentration of clearance presented in ‘Nuclear Safety Commission Notice No. 2020-6’. Therefore, this study is expected to be able to flexibly cope with large amounts of radioactive metal waste generated during the nuclear power plant operation/decomposition by completing verification and applicability review of the ‘Nanoparticle Complex-fluid Decontamination Device’ for the decontamination of VLL radioactive metal waste.

Keywords: Nanoparticle complex-fluid decontamination technology, Waste liquid treatment technology, Nanoparticle complex-fluid decontamination device, Very low level, Radioactive metal waste

Assessment of Diffusional Release in IAEA ISAM Engineered Vault Disposal Facility

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Regarding the performance and/or safety assessment of radioactive waste disposal facilities, it is essential to develop assessment models that are more detail and reflects reality. The major key to enhance safety margins is to reduce unnecessary conservativeness by performing the realistic safety assessment.

In this paper, a diffusional release source-term model is implemented to engineered vault type disposal facility using GoldSim to investigate the effect of diffusion behavior in the cement-solidified waste forms against dissolution of radionuclides in the waste forms either instantly or gradually. In GoldSim, the source, cell, and pipe pathways are used to transport the radionuclides from the waste forms to biosphere and doses of human receptors are calculated according to the definitions that user defines. Firstly, the whole disposal system in IAEA ISAM Vault Test Case was modeled by GoldSim and compared the results with the those of the previous AMBER modeling which has non-diffusional source-terms. Then, GoldSim diffusional model is implemented in the source portion. To model the diffusional release source-term model, waste forms, backfills, and vaults were divided into a series of concentric rectangular parallelepiped cells and the diffusive mass fluxes between neighboring cells were calculated using diffusive mass flux links based on Fick's First Law. Also, infiltration rate of water, waste drum failures and degradations of concrete and vault materials were applied to source-term model. Diffusion of radionuclides occurs through a concentration gradient which expressed as the change of concentration caused by a change of position and the local diffusional flux is defined with diffusivity and concentration gradients. In liquid release scenario, radionuclides will diffuse in source material, released from the vault, transport to well through unsaturated and saturated zones via advection of ground-water flow. The peak concentrations of the key radionuclides (C-14, Tc-99, I-129, U-234) and their timings in drinking water from the well were calculated and compared with the those of the previous study.

As the result, the diffusional release source-term model shows lower peak concentrations and later timings than non-diffusional release model, and radionuclides which have relatively low half-life show more significant differences. As effective diffusion coefficients were decreased, radionuclides peak times were delayed and peak concentrations became smaller. When the nuclides-specific effective diffusion coefficients in the cement-solidified waste forms were obtained, it is expected that more safety margins will be obtained for post-closure safety assessment of the engineered vault type disposal facility.

Keywords: IAEA ISAM engineered vault test case, Safety assessment, Diffusional release, GoldSim

Study on the Reduction of Tributyl Phosphate (TBP) Using Underwater Plasma

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Tributyl phosphate (TBP) is used in large quantities (3,000–5,000 Ton/annum) as an extractant for uranium and plutonium in nuclear fuel reprocessing facilities. In addition, it has various industrial uses such as defoamer, plasticizer, herbicide and hydraulic oil. TBP containing aqueous waste is generated in large quantities due to its widespread use, and depending on the application, TBP waste contains radioactive waste. Regardless of the presence of radioactive waste, TBP should be treated to prevent environmental contamination. TBP is a very stable compound, and it has a characteristic that it remains for a very long time in environments such as soil and water. As a result, environmental pollution caused by TBP is continuously detected in indoor air, water quality environment, influent and effluent from wastewater treatment plants in domestic and industrial sites, and thus needs to be treated. Chemical decomposition, biological decomposition, incineration, or immobilization methods have been considered as existing technologies for TBP treatment, but a new treatment technology is required due to environmental and economic problems.

Advanced oxidation process (AOP) is suitable for processing high-concentration organic waste based on the high oxidizing power of OH radicals, but the existing AOP technology is not suitable for large-capacity treatment due to problems of efficiency or economic feasibility. The plasma-based AOP technology used in this study can be designed simply and economically, and it is advantageous for generating OH radicals because H_2O_2 , O_3 , and UV are simultaneously generated. The electrical and optical properties of the underwater plasma device were analyzed, and the change form of the solution according to the plasma treatment time and the change in the properties of the solution were described. In addition, the reduction of total organic carbon (TOC) in TBP was analyzed to confirm the reduction of organic matter in TBP.

Acknowledgements

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Keywords: Underwater plasma, OH radical, Tributyl phosphate, Plasma based AOP

Effects of Gamma Irradiation on Thermal Stability of Cation Ion Exchange Resin

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The gamma irradiation effects of thermal decomposition on cation ion exchange resin (IRN77) were investigated. Gamma irradiation was performed with doses of 0, 300, 500 and 700 kGy. To evaluate resin decomposition in radioactive waste disposal site, thermal gravity analysis of irradiated ion exchange resins was achieved under air and nitrogen conditions. The complete oxidative decomposition (residue within 1%) was achieved over 700°C, but about 30% residue of resin was observed under nitrogen conditions. The onset temperature of sulfon group decreased with irradiation dose at 0, 300, 500 and 700 kGy was about 275, 265, 260, 255°C irrespective of air supply. This shows the easy decomposition of irradiated ion exchange resin. The risk and possibility of radioisotopes release should be thoroughly evaluated because of easy decomposition of sulfon group by gamma irradiation.

Keywords: Cation exchange resin, Decomposition, Gamma irradiation, Onset temperature

Considerations for the Treatment Methods of Radioactive Waste Generated When Replacing the Nuclear Reactor Head (RRVCH) in Korea

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In Korea, reactor head replacement (Replacement Reactor Vessel Closure Head, RRVCH) has been successfully replaced, starting with Kori Unit 1, Hanbit Units 3 and 4, and Kori Unit 2. In the near future, the ORVH (Old Reactor Vessel Head) of Hanul Units 1 and 2 will be replaced. In this study, the treatment method was studied based on the experience for treatment method of radioactive waste generated during RRVCH. The main wastes types are divided into ORVH, CRDM (Control Rod Drive Mechanism), coil stack, stud bolt, vent-line, dummy can, seismic plate, etc.

The treatment method of radioactive waste generated during the replacement of the reactor head was implemented based on the ALARA (As Low As Reasonably Achievable) concept, which is reasonable for radioactive waste reduction, treatment and exposure optimization when replacing the reactor head.

In addition, to identify the optimal working conditions and minimize problems to minimize radiation exposure, and achieve ALARA through mock-up training.

Currently, 24 nuclear reactors are in operation in Korea, and 2 nuclear reactors are permanently shut down. Of these, 9 nuclear reactors that have been in operation for more than 30 years (as of September 2021). If the radioactive waste management and treatment method applied to the replacement of the reactor head performed previously with domestic technology is applied, it is expected that the ALARA will be achieved and the cost of waste management will be reduced.

Keywords: RRVCH, ORVH, Radioactive waste, CRDM, Vent line

Analysis of the Liquid Radioactive Waste Generation in HANARO

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This study is about the contents of analysis of nuclide in liquid waste generated during the normal operation of the HANARO Research Reactor in 2013–2014. Sump of the liquid waste system is a facility where all liquid wastes generated in the reactor building are collected, and liquid wastes are mainly generated when facility maintenance works such as replacing filters and ion exchange resins in the purification system. The water level of the sump increases even when a water leak occurs due to a system malfunction or pipe breakage, but this problem did not occur during the 2013–2014 period.

Reactor operators collect samples of this sump every quarter and check whether radioactive materials show radioactivity levels within the allowable range. Check that the alpha, beta, gamma, and tritium values are within the limit of $3.7 \text{ kBq}\cdot\text{cm}^{-3}$ through the sample. Among gamma-emitting nuclide, Co-60 and Sb-124, which have long half-lives, are the main targets of management, and tritium, which may cause internal exposure, is also importantly managed.

As a result of analyzing the average radioactivity concentration during the 2013–2014 period, alpha rays indicated within the measurement range. Beta-ray showed 0.08% of the reference value and tritium 2.2%, Co-60, a major gamma-emitting nuclide, showed 0.16%, and Sb-124 showed 0.003%.

Keywords: HANARO, Liquid waste, Radioactivity, Tritium

Analysis of Waste Filters and Waste Resins Generated From HANARO

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The HANARO is a multipurpose research reactor that has been operated since 1995. In HANARO, demineralized water is used as a coolant for reactors and spent fuel. It is important to maintain demineralized water quality above a certain level to prevent corrosion of reactor structures and fuel and to reduce radioactive products in the coolant to keep radioactivity levels low.

For such water quality management, a purification system is included in the primary cooling system and the spent fuel cooling system. This purification system consists of filters and ion exchange resins. Large particles of foreign substances or debris are filtered out through the filter, and radionuclides are filtered out using an ion exchange resin that uses zwitterions.

The filter is designed to be replaced when the differential pressure of filter exceeds a certain level due to foreign substances caught in the filter. The ion exchange resin is designed to be replaced when the electric conductivity of the coolant exceeds a certain value. These waste filters and waste resins are classified as low-intermediate-level radioactive waste and stored in the temporary storage warehouse of the reactor building. This paper analyzed the amount of waste filter and waste resin, one of the types of solid radioactive waste generated in HANARO, and predicted the amount of waste generated annually.

Keywords: Research reactor, HANARO, Purification system, Filter, Ion exchange resin, Solid radioactive waste

Review on the Solubility Behavior of U(IV) in EDTA and ISA Containing Aqueous Solution

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KORAD received and disposed of about 27,000 drums of low- and intermediate-level radioactive waste from 2014 to August 31, 2021. This solid waste material must satisfy acceptance criteria in various tests such as compressive strength test, immersion test, and leaching test, and the total amount of radioactivity contained in the waste package should be limited. In particular, long-lived radionuclides such as H-3, C-14, and alpha-emitting radionuclides can only be disposed of as waste with less than the allowable radioactivity. This is to prevent migration of radionuclides into the environment during long-term storage for more than 5,000 years.

Cellulose contained in wood, paper, etc. is degraded under the condition of pore water in the cement-solidified material to form ISA, which forms a complex with radionuclides, and increases the solubility of radionuclides. Therefore, even if the solidified radioactive waste satisfies the leaching/immersion test before disposal, the ISA generated during long-term storage forms a complex with radionuclides, thereby reducing the safety of the disposal facility. Therefore, it is necessary to study systematically the complex formation mechanism between radionuclides and organic substances such as ISA, and the phenomenon of increasing solubility by complexes.

In this study, the speciation and solubility behavior of U(IV) ions in aqueous solutions containing EDTA and ISA were investigated. After collecting thermodynamic data from literature, optimal data were selected and ionic strength were compensated to fit for MINEQL program. The speciation of each ionic species and complex species was calculated through the MINEQL program. The critical ISA concentration affecting the solubility of U(IV) was determined and the effect of U(IV)-ISA or U(IV)-EDTA complex species on the solubility was investigated.

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Keywords: Radionuclide waste, Disposal facility, Uranium, Solubility, ISA, EDTA

Review of Maritime Radioactive Waste Transportation Risk Assessment Codes

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As nuclear power plants are operating 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. In order to transport radioactive wastes, risk assessment for radioactive waste transportation should be preceded. The object of this study is to investigate and analyze maritime radioactive waste transportation risk assessment codes.

In this study, MARINRAD, POSEIDON, and LAMER were investigated. MARINRAD was developed by Sandia National Laboratory. MARINRAD evaluates risk by calculating concentrations of radionuclides in ocean, concentration factors for marine organisms, and doses by exposure pathways. MARINRAD uses compartment modeling to calculate concentrations of radionuclides in ocean. Concentration factors of marine organisms are calculated through food chain modeling assuming a steady state. Based on the derived concentrations of radionuclides and the concentration factors of marine organisms, risk is evaluated by calculating doses for external exposure, ingestion, and inhalation. POSEIDON was developed by the CEPN in France. POSEIDON evaluates risk by calculating concentrations of radionuclides in ocean, concentrations of radionuclides in marine organisms and doses by exposure pathways. POSEIDON uses the box modeling method when calculating concentrations of radionuclides in ocean. The concentrations of radionuclides in marine organisms are calculated using static or dynamic food chain modeling based on the derived concentrations of radionuclides. Based on the derived concentrations of radionuclides in ocean and marine organisms, risk is evaluated by calculating doses for external exposure, ingestion, and inhalation. LAMER was developed by the JAEA. LAMER evaluates risk by calculating concentrations of radionuclides in ocean and dose for the intake of marine organisms. LAMER derives concentrations of radionuclides by dividing the global oceans at 2° intervals for latitude and longitude, respectively. Based on the derived concentrations of radionuclides, risk is evaluated by calculating dose for ingestion of marine organisms considering concentration factors of marine organisms, dose conversion factors, catch, and population.

Maritime radioactive waste transportation risk assessment codes investigated in this study commonly include a process of calculating concentrations of radionuclides in ocean by dividing ocean into multiple zones and a dose calculation process based on ingestion of marine organisms. The results of this study can be used as basis for risk when transporting radioactive wastes in Korea.

Acknowledgements

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Keywords: Radioactive waste, Maritime transportation, Risk assessment, MARINRAD, POSEIDON, LAMER

Study of Treatment Possibility Towards Decommissioning Waste Using HIP

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The R&D of technologies for treating efficiently various and large quantity of decommissioning wastes, which are produced during decommissioning of nuclear power plants, are proceeding actively over the world. Especially the technologies for reduction of waste volume are welcomed and these are super compaction, melting, thermochemical treatment, etc. to meet the type and characteristics of wastes.

The combined technology of sintering and compaction technology, namely HIP (Hot Isostatic Pressing) is the technology developed in 1953 firstly by ASEA for diamond processing and is the technology to press the target isostatically by using inert gas at about 1,000°C and 100–200 MPa (about 1,000–2,000 atm), and is well welcomed recently in Japan and England, etc. in the field of radioactive waste treatment because of high compaction capability.

Thus this paper investigates the various cases using HIP technology and suggests the application possibility for the reduction of volume of local decommissioning wastes.

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Keywords: Hot isostatic pressing, Decommissioning waste, Sintering, Inert gas

The Assessment of Exposure Dose of Radiation Worker for Decommissioning Waste in the 2nd Phase Disposal Facility

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In the case of decommissioning a nuclear power plant that generates various types of waste, a new disposal container is necessary in consideration of the characteristics of radioactive waste, the specification of the disposal facility, the convenience of operation, and the safety of disposal. So various low and intermediate-level waste disposal containers are being developed in KORAD. Also an assessment of disposability with new disposal containers is required. These new containers to be disposed at the second phase disposal facility must have an exposure dose lower than $0.02 \text{ mSv}\cdot\text{h}^{-1}$ for remote worker and $0.001 \text{ mSv}\cdot\text{h}^{-1}$ for worker near a sealed repository according to the exposure dose limit in the safety analysis report.

In this study, 5 normal exposure scenarios and 4 abnormal exposure scenarios were derived for the assessment of exposure dose of worker during the operation of the second phase disposal facility. And the exposure dose of worker in each scenario was evaluated using the MCNP code. As a result, the exposure dose rates in all normal scenarios were calculated to be less than the worker direct exposure limit of the safety analysis report. In the case of abnormal scenarios, the working hours that do not exceed $20 \text{ mSv}\cdot\text{y}^{-1}$, the annual dose limit for radiation worker, was calculated. The results of this study will be useful in establishing the optimal radiation working conditions in disposal facility.

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Keywords: Decommissioning radioactive waste, Radiation worker, Disposal container, Exposure dose

Grouping Strategy for Packaging Radioactive Waste Filter Drum

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Currently, radioactive waste filters are stored at temporary spaces in nuclear power plants for a long time due to decay their radioactivity. Using a robot system that can be remotely controlled in a highly radioactive environment, the radioactive waste filter is safely withdrawn with classified by the radiation dose rate. In order for the robot system to withdraw waste filters from the filter room and pack radioactive waste drums through compression and packaging systems, it is necessary to consider whether the surface dose and the other acceptance criteria for the radioactive waste disposal are met. For an efficient treatment process, it is important to classify and group the radioactive waste filters by surface dose rate. Therefore, the surface dose rates of filters in nuclear power plant were investigated and it was found that the reduction ratio of the metallic part of filters is about 70% based on the compressive strength of 100 tons. Therefore, the following classification strategies were established in consideration of the requirements for compression and disposal acceptance criteria.

- Low-dose radioactive waste filters are to compressed and packaged in a non-shielding drum, and three or more are compressed until the surface dose is 2 mSv/hr.
- Medium-dose waste filters are to compress and packaged in the shielding drum, and two or more are compressed until the surface dose is 2 mSv/hr.
- High-dose waste filters are packaged without compression in the shield drum until the surface dose is 10 mSv/hr for the disposal of the rock cavern facility.

Based on these grouping strategies by surface dose rate of radioactive waste filters, the cost and safety of disposal of radioactive waste filters will be satisfied. In addition, it will be possible to minimize the exposure dose of workers by efficient handling of radioactive waste filters by remote control robot devices.

Keywords: Radioactive waste, Waste filter, Surface dose, Grouping strategy

Conceptual Design of Compression and Packaging System of Radioactive Waste Filters

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Radioactive waste filters temporarily stored in nuclear power plants have various surface dose rate, resulting in effective compression and packaging methods. Establishing reliable compressive and packing system is needed for packaging drums to ensure cost-benefit effects and disposal suitability. Radioactive waste filters grouped and withdrawn from the filter room by the robot system are delivered to the conveyor and elevator movement system, and the surface dose is precisely measured at the top of the elevator, and then to the waste filter fixed cask of the compression and packaging system. To compress and pack the waste filters for safe handling of radioactive waste, the waste filter compression and packaging system was designed with the following concepts.

- Open the moving disk of the elevator system near the upper part of the waste filter fixing cask and insert the radioactive waste filter into the fixed cask.
- Move the fixed cask to the compressed position.
- The compressed piston compresses the radioactive waste filter according to the preset compression ratio.
- The compressed waste filter is moved by the portable tongs and inserted in a packaging waste drum.
- The packaging drum will be transferred to disposal facility after satisfying the disposal acceptance criteria.

By developing a compression and packaging system with robot system, a safe and efficient treatment technology for high-dose radioactive waste filters on the nuclear power plants will be secured.

Keywords: Radioactive waste, Waste filter, Compressive and packing system, Radioactive waste drum

The Review of Release From Spent Fuel Pool of Indian Point Nuclear Power Plant

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In 1992, a small white radioactive precipitate was discovered above the ground surface on the outside of the Indian Point unit 2 spent fuel pool (IP2-SFP) east concrete wall. The radiological characteristics of the precipitated material were consistent with a potential leak from the SFP. The investigation to the SFP was conducted by the divers and they found indications that the liner had been gouged when an internal rack had been removed in 1990. And two hundred and forty linear feet of IP2-SFP wall welds were inspected and vacuum-tested. As a result, the damaged liner was confirmed to this only one case and repaired.

During construction excavation in 2005 for dry cask storage, the outside wall of IP2-SFP was exposed and two shrinkage cracks were founded. Moisture was seen in these cracks then subsequently covered with a plastic membrane to prevent moisture evaporation and collect adequate liquid volume samples for analysis. As a response of this event, the accessible areas of the IP2-SFP were inspected and six suspect areas coated with epoxy.

The combined average of release rate from two cracks was high as 2 l/day in initially. However, the release rate had fallen off rapidly until 2007 and become intermittent approximately 0.02 l/day (0.83 cc/hr), when flowing. The concentration of tritium in the moisture was analyzed 810,000 Bq·L⁻¹ in maximum and continuously decreased to 110,000 Bq·L⁻¹ until 2007. And also the concentration of tritium of groundwater monitoring well located near the IP2 was shown 5,213 Bq·L⁻¹ in 2005.

During the inspection period, the IP2-SFP transfer canal was drained and vacuum box testing of the welds was conducted in 2007. Through the test a single small imperfection in one of the liner plate welds was discovered and repaired.

Although there were some imperfection of liner plate welds and leak from the SFP, it was confirmed that the level of radioactivity did not affect the residents.

Keywords: Unplanned release, Groundwater, Spent fuel pool, Radionuclide

Effect of Humic Acid on Iodine Speciation and Removal Using Layered Double Hydroxides

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Synthetic nanostructured layered double hydroxides (LDH) are promising sorbents for ^{129}I removal. According to a few previous studies, the dissolved humic acid reduces the iodate (IO_3^-) resulting in the formation of organoiodine (Org-I). Therefore, this study investigates the effect of humic acid on the iodine speciation in groundwater and the resulting effect on the removal of iodate by layered double hydroxides. The co-precipitation method was utilized for the synthesis of LDH comprised of Co and Cr (CoCr LDH) or Cu and Al (CuAl LDH). Moreover, the engineering scale pellets of LDHs were fabricated with the polysulfone-alginate mixed binder for dynamic column experiments. The batch sorption tests and column experiments were conducted with varying amounts of humic acids ($0\text{--}35\text{ mg}\cdot\text{L}^{-1}$) in iodate solution ($100\text{ mg}\cdot\text{L}^{-1}$). Iodine species were analyzed with inductively coupled plasma mass spectrometry (ICP-MS), ion chromatography (IC), and total organic carbon (TOC). In a control sample without any solids, 30% of iodate was changed to Org-I in presence of humic acid ($35\text{ mg}\cdot\text{L}^{-1}$). The humic acid facilitated iodate reduction to Org-I decreased the iodine removal capacity of CoCr LDH ($\sim 27\%$ lower in iodate removal efficiency), increasing the mobility of iodine in the environment. On the other hand, unlike the CoCr LDH, the iodate removal efficiency of CuAl LDH was still 99% even in presence of humic acid. Moreover, the Org-I was efficiently removed up to 95% by CuAl LDH.

Acknowledgements

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Keywords: Organoiodine, Natural organic matter (NOM), Layered double hydroxides, Engineering scale pelletization of LDHs

Review on the Reasonable Methodology for Indirect Evaluation of Difficult-to-Measure Nuclides Inventory in Radioactive Waste Packages

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In order to dispose of radioactive wastes, it is necessary to satisfy not only physicochemical disposal suitability but also radiological suitability. For this, a reliable evaluation of difficult-to-measure (DTM) nuclide inventory is required for radioactive waste packages. The scaling-factor method was developed about 12 years ago and applied as an indirect evaluation method for DTM nuclide inventory based on economic feasibility and efficiency for radiochemical analysis of a large amount of radioactive waste packages. Scaling-factors initially developed in Korea were applied immoderately due to their high dependence, and they tend to be determined too conservatively because of concerns about underestimation and uncertainty. Since this may impair the efficiency of the repository, excessive conservatism should be avoided by lowering uncertainty and increasing accuracy in consideration of the domestic situation where disposal conditions are not sufficient. First, it is necessary to establish a systematic representative sampling procedure that broadly reflects the generation and radiological characteristics of the population to be evaluated for indirect evaluation of DTM nuclide inventory. Second, it is important to select an appropriate method for indirect evaluation of DTM nuclide and statistical techniques. For this purpose, an accurate statistical understanding of each evaluation method is required. As typical indirect evaluation methods, there are the scaling-factor method and the mean radioactivity concentration method. Although it is somewhat different for each country, the scaling-factor method is applied when there is a significant correlation between the key nuclide and DTM nuclide, and when there is no or insufficient correlation, an alternative method such as the mean radioactivity concentration method is applied. Third, it is necessary to determine the statistical processing methods and statistical factors suitable for the selected indirect evaluation method in consideration of realistic conditions such as domestic disposal conditions, regulatory environment, and technological level. Therefore, we reviewed the improvement direction of the indirect evaluation for DTM nuclide inventory in radioactive waste package such as the method calculate the scaling-factor according to the linearity of the scatter plot under the conditions of significant correlation with the systematic representative sampling method of domestic nuclear power plant (geometric mean, linear regression of logarithms), the method to calculate the reasonable conservative scaling-factor when an uncertainty (confidence limit) exceeds the allowable uncertainty at 95% confidence level, the method to calculate the minimum number of analysis data to satisfy the confidence limit of scaling-factor.

Keywords: Scaling-factor, Indirect evaluation, Nuclides inventory, Statistical technique

The Origin and Role of the Groundwater Protection Initiative in the US

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At least 14 events over a period of ten years (from 1996 to 2006) of unplanned releases to soil and or groundwater had been occurred in the US. Most of the unplanned releases events had been occurred at spent fuel pool and radiological waste line. Although there was unplanned release to groundwater, both the Nuclear Regulatory Commission (NRC) and Nuclear Energy Institute (NEI) agree there was not a threat to public health and safety. The NEI is a nuclear industry trade organization representing its member companies in the United States. And the NEI initiative is a set of industry actions undertaken collectively for the benefit of the members.

However, at that time, the US nuclear industry was anticipating a nuclear renaissance with about 15 new nuclear power plants in some state or planning. The groundwater issues were not helpful at a national policy to the anticipated new proposed NPPs. Thus the guidance document “Industry groundwater protection initiative” NEI 07-07 is issued in 2007. And minor revision of the document was issued in 2019.

Since the NEI 07-07 represented a commitment on the part of the NEI member companies. NEI 07-07 is policy document or commitment by each company. And all NPPs, in the US, use NEI 07-07 as the basis for their individual groundwater protection programs.

Keywords: NEI 07-07, Groundwater protection, Nuclear power plant, Unplanned release

Regulatory Clearance Process of Soil Radioactive Waste in 2019 at KAERI

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In 2019, Korea Atomic Energy Research Institute (KAERI) cleared 67.3 tons of soil radioactive waste. It was generated from 2013 to 2015 at KAERI, during constructing the nuclear facility nearby the space dealing with uranium ore previously. The clearance process was as follows: collect samples, analyze radioactive concentration, assess exposure dose, get permission from the regulatory body in Korea, transport, dispose and monitor the waste. The greenhouse was used to store soil and collect samples safely. To characterize soil, samples were analyzed by using Energy Dispersive X-ray Fluorescence (ED-XRF). As a result, the major radionuclide was U-238, and the maximum radioactivity concentration was $7.34 \times 10^{-2} \text{ Bq} \cdot \text{g}^{-1}$. The value was used to perform RESidual RADioactivity (RESRAD) code, and the exposure dose was assessed as individually $7.77 \times 10^{-1} \text{ uSv/year}$ and collectively $1.37 \times 10^{-2} \text{ man} \cdot \text{Sv/year}$. It meets the criteria set by the Atomic Energy Act of Korea, which is less than 10 uSv/year for individual dose and $1 \text{ man} \cdot \text{Sv/year}$ for collective dose. Clearance was permitted by Korea Institute of Nuclear Safety (KINS). After permission, the soil was transported to the committed disposal area. Surface radiation dose rates of the landfill and the waste were similar. Transportation was carried out four times, and the radiation dose rates on each truck were between 0.8 and 0.10 uSv/hour. Radiation dose rate at the site was measured before and after disposal, and the values were 0.06 uSv/hour and 0.10 uSv/hour each. The environmental dose in Korea is between 0.05 and 0.30 uSv/hour. As a result of the clearance process, radioactive waste disposal costs saved, and waste management efficiency increased due to the reduction in the amount of waste to be managed.

Keywords: Regulatory clearance, Radioactive waste, Clearance process, Soil radwaste

Study for Ease of Decommissioning on New Radioactive Waste Treatment Facility

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KAERI has a plan to construct the new radioactive waste treatment facility for Very Low Level Waste and Low Level Waste. The Nuclear Safety and Security Commission has legislated about the permission of preliminary decommissioning plan for radioactive waste treatment facility in Jun, 2021. Accordingly, willing licensee should prepare decommissioning plan in chapter 5 of Safety Analysis Report (SAR). Ease of Decommissioning is evaluated by NRC Regulatory Guide 4.21 for over the whole life cycle of a facility. This Guide was composed 4 sub-sections: 1) Minimizing Facility Contamination, 2) Minimizing Contamination of the Environment, 3) Facilitating Decommissioning, 4) Minimizing the Generation of Waste. Based on this guide, ease of decommissioning of new radioactive waste facility was analyzed.

The new radioactive waste treatment facility confirmed to reflect in designing the facility including pressure differential, radiation shielding, leakage prevention, which is necessary to prevent environmental hazard and to minimize radioactive waste during the operation and decommissioning. This analysis is a very important task as it confirms the safety of nuclear facilities, and it is a matter to be reflected every time when designing new facilities.

Keywords: Decommissioning plan, Ease of decommissioning, Design properties, Radioactive waste treatment facility

Development on Dose Assessment Code for Decommissioning Waste Recycling and Disposal

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In Korea, various types of decommissioning wastes are classified according to the criteria of radioactive concentration in the Nuclear Safety and Security Commission Notice No. 2020-6. The characteristic of decommissioning wastes is the large quantity of metal and concrete wastes with low activity, most of which are less than the allowable concentration for disposal. In abroad, disposal or recycling of this kind of wastes are widely and actively carried out.

Currently, the disposal facility capable of disposing 100,000 drums based on 200 L is in operation for radioactive waste disposal, and the second phase facility capable of disposing 125,000 drums is currently under construction. Due to the limited storage capacity of the disposal facility, reduction of decommissioning waste through recycling and disposal is highly required for the efficient operation of the disposal facilities.

This study is to develop a scenario for disposal and recycling of decommissioning waste such as metal and concrete generated in large quantities, and to develop a computational code for dose assessment from each recycling and disposal scenarios. The computational code is aimed to calculate external and internal doses by simulating the exposure pathways from waste generation to transport, processing, manufacture, and final destination of recycling or disposal.

The input parameters of dose assessment code will be derived for the exposure target of each exposure pathways reflecting domestic characteristics. And the dose conversion factors will be calculated based on these factors, and will be reflected in the calculation process. The input parameter are composed of concentrations of each radioactive nuclide at the decommissioning site and the time of each exposure scenario, and the output shows the dose results of the exposure target corresponding to each.

As a result of this study, we will develop scenarios for disposal and recycling of large amounts of decommissioning wastes, a dose assessment code for the safety evaluation, and technical criteria on the radioactive nuclide concentrations for disposal or recycling of decommissioning waste. End point of this study is to develop the manual of decommissioning wastes management for the reduction of overall cost of decommissioning wastes, to present quantitative technical criteria which will be possible to raise social awareness and public acceptance of decommissioning wastes, and to economically operate radioactive waste disposal facility.

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Keywords: Nuclear decommissioning, Decommissioning waste, Recycling, Disposal, Dose assessment code

Development of a Dose Assessment Computational Model for Both Internal and External Exposure in Clearance of Metallic and Concrete Radioactive Waste Recycling

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Large amounts of radioactive waste are generated during decommissioning of NPPs, of which 67.1% are VLLW, and most of them are metallic/concrete waste. To reduce the decommissioning cost and efficiently use disposal facilities, it is necessary to utilize the clearance system for low contaminated metallic/concrete waste. For the clearance of radioactive waste, individual doses must satisfy $10 \mu\text{Sv}\cdot\text{y}^{-1}$ for specific scenarios. Thus, the dose assessment code suitable for the recycling scenarios is required. The commercial code currently used for clearance of radioactive waste need to be improved in the following aspects. First, the domestic recycling industry status should be reflected. Second, the various geometries of the source, the shielding effect, and various types of exposure must be considered. Third, metallic/concrete radioactive waste recycling scenarios can be simulated.

The new dose assessment code for the clearance of metallic/concrete radioactive waste have been developed by the Ulsan National Institute of Science and Technology (UNIST). This paper shows a validity of the external dose calculation module for various source geometries simulating metallic/concrete recycling scenarios and the internal dose calculation module applied the internal dose assessment model of the commercial code. The external dose calculation module uses the Point Kernel method to simulate for various geometries, which are drums, buildings, and ingots assuming the metallic/concrete waste recycling procedure. For verification of external dose calculation module, this research conducts a verification with results of the RESRAD-RECYCLE designed for evaluating radiation exposures resulting from recycle of contaminated metals and the best-known shielding code, MicroShield®. And for the internal dose assessment, this paper compared and verified with the RESRAD-RECYCLE currently used in dose assessment for the clearance.

In the case of external exposure, this research conducts a verification with RESRAD-RECYCLE and MicroShield® for two gamma emitters, Co-60 and Cs-137. The result of RESRAD-RECYCLE shows the slight difference about 10 to 20%. And it compared MicroShield® under four geometries, which are disc, line, cuboid, and sphere. It also shows the slight difference of up to 10%. For the internal exposure case, it compared RESRAD-RECYCLE for beta emitters and almost similar results were obtained with a ratio of 0.997 to 1.00. Therefore, developed dose assessment module has a validity compared to the existing codes. Further research will be conducted to evaluate the validity of dose assessment considering a shielding effect and concrete waste recycling scenarios. The research about domestic scenarios for metallic/concrete waste recycling will also be conducted.

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Keywords: Radioactive waste, Dose assessment tool, Recycle, Clearance, Decommissioning

Solidification of Spent Ion-exchange Resin Using Metakaolin-based Geopolymer Waste Form

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Nuclear grade ion-exchange resins (IERS) are used for removing the radionuclides such as ^{90}Sr , ^{60}Co , ^{137}Cs , ^{235}U , and etc. in radioactive liquid waste. The spent IERS are classified as low- and intermediate-level radioactive wastes (LILW) and should be immobilized in order to decrease the potential danger to environments before disposal in repository. Cement solidification is conventional to solidify spent IERS because of advantages such as reasonable cost, simple process, and various applications. However, cement waste form has a porous microstructure and is prone to structural degradation under acidic conditions or high temperatures. In addition, several problems such as low waste loading, crack, swelling, long curing time, and low leaching resistance in cement waste form have been reported. Thus, using cement waste form for solidification of spent IERS containing various radionuclides is very challenging.

As an alternative solidification method, geopolymer which can be prepared at room or low temperatures with relatively fast curing process is a promising inorganic binder. Geopolymers can be synthesized by combining aluminosilicate materials such as metakaolin or coal ash with alkaline activator. They have good physicochemical properties such as early-age compressive strength, acidic resistance, thermal stability, and capacity to immobilize radionuclides when compared to cement waste form.

In this study, simulated spent IERS were tested for solidification using metakaolin-based geopolymer waste form. The $\text{K}_2\text{O}:\text{Al}_2\text{O}_3:\text{SiO}_2:\text{H}_2\text{O}$ ratio of 1:1:4:12 was maintained using metakaolin and potassium silicate solution in geopolymer formulations. For enhancement of mechanical strength, 1wt% of $\text{Ca}(\text{OH})_2$ was used as a chemical additive. IRN-150 was used to prepare the simulated spent IER 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 $\text{mg}\cdot\text{g}^{-1}$ -dried IER, respectively).

The 7-d compressive strength of the geopolymer containing 20wt% simulated spent IERS was 2 MPa, while the geopolymer containing 1wt% $\text{Ca}(\text{OH})_2$ under the same condition showed a higher result (6 MPa). However, both geopolymers were degraded during the immersion test because the dried resin was swollen when it reacted. To prevent this problem, simulated spent IER was immersed into the water to be fully saturated and used for the geopolymer solidification. In addition, the liquid/solid ratio of geopolymer was kept constant as 1.0 to enhance the water resistance. Consequently, geopolymer was not degraded during the immersion test and had similar compressive strength (6 MPa). This study provides insight for the geopolymer solidification process and optimized formulation for disposal of spent IERS.

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Keywords: LILW, Spent ion-exchange resin, Solidification, Geopolymer, Metakaolin

Compressive Strength of Polymer-cement Waste Form for Solidification of Liquid Radioactive Waste From Decommissioning of NPPs

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There has been receiving a great interest to manage liquid radioactive waste generated from decommissioning of nuclear power plants. Cement has typically been used to solidify radioactive liquid waste. Synthetic polymer can be introduced as an additional aggregation reagent to enhance physical properties such as compressive strength. In this study, we prepared a type of polymer-cement, where the epoxy resin was added to the cement with a weight ratio of 4.5:2, as a solidifying agent. Using the agent, a waste form was produced by incorporating zeolite particles which contain simulated waste (with a weight ratio of 1:1). A series of tests including immersion, irradiation, and thermal cycle tests were performed on the waste form. Each test was performed in accordance with ANSI/ANS 16.1, US NRC Technical Position on Waste Form, Rev. 1, ASTM B 553, respectively, followed by a compressive strength test in accordance with KS F 2405. Each test used 3 specimens for measurement and came up with an average value of compressive strength.

The initial compressive strength of the waste form as prepared was found 3,244 psi. The compressive strength after immersion of the waste form in water for 90 days showed 1,716 psi, resulting in 47% decrease compared to the initial strength. An irradiation with the waste form up to 1.0×10^9 Rad gave the strength as 2,306 psi, resulting in 29% decrease. The compressive strength after 8 thermal cycle test with the waste form at a temperature range of -40°C to $+60^\circ\text{C}$ was 1,895 psi, resulting in 42% decrease. In spite of the thermal effect, cracks did not occur. Even with such amounts of decrease, the compressive strength maintained more than 3 times higher than the disposal site acceptance criteria (500 psi).

In summary, the proposed polymer-cement was confirmed highly acceptable in terms of compressive strength, and expected to be a good solidifying agent for radioactive liquid waste contributing to a sufficient structural stability for the disposal on the low- and intermediate-level radioactive waste repository of KORAD.

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Keywords: Polymer-cement, Compressive strength, Immersion test, Irradiation test, Thermal cycle test

Geopolymer Synthesis by Recycling Calcium Hydroxide Obtained From Nuclear Power Plant Dismantling Waste

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In the nuclear industry, the dismantling of nuclear facilities is essential as part of the nuclear facility cycle. Radioactive waste generated during the decommissioning of nuclear facilities must comply with strict regulatory standards, resulting in a large amount of polluted waste. About 40% of the cost of dismantling a nuclear power plant is the waste treatment cost incurred during the dismantling process, of which concrete waste accounts for a significant portion of the total waste. By replacing the radioactive waste generated during the decommissioning process with waste treatment material, it is possible to reduce the amount of radioactive waste generated and the cost of treatment.

Geopolymers are inorganic binder material formed by the reaction of amorphous aluminosilicate materials with alkali activator, and is being studied for solidification of radioactive waste. In the cement powder of concrete, Ca^{2+} is present, which can be used as alkali activator in the synthesis of geopolymers. The use of Ca^{2+} present in concrete waste to replace geopolymer materials can significantly reduce the amount and disposal costs of radioactive waste generated during operation and decommissioning of nuclear power plants. The goal of this study is to prepare $\text{Ca}(\text{OH})_2$ using Ca^{2+} extracted from concrete waste instead of conventional alkali activators (NaOH , KOH , etc.), and to synthesize a geopolymer.

For the production of $\text{Ca}(\text{OH})_2$, the metal in the waste concrete that crushed to less than 30 mm was removed, and hydrochloric acid was added to dissolve it. $\text{Ca}(\text{OH})_2$ was precipitated and recovered by adding NaOH to the calcium compound (CaCl_2) separated in a liquid phase by reacting with hydrochloric acid. The resulting $\text{Ca}(\text{OH})_2$ was added to the geopolymer in a ratio of 5–15 % after characterization, and its performance was evaluated through compressive strength and leaching tests.

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Keywords: Recycling calcium hydroxide, Geopolymer, Concrete waste, Nuclear power plants dismantling

A Study on Evaluation of Leachability Index Using ANSI/ANS-16.1

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The Korea Radioactive Waste Agency (KORAD) requires identification of various properties of solidified waste, usually specified in Waste Acceptance Criteria (WAC), for the disposal of them. The primary properties are leachability index, radiation exposure resistance, compressive strength, etc. The leachability index, obtained using ANSI/ANS-16.1, Measurement of the Leachability of Solidified Low-Level Radioactive Wastes by a Short-Term Test Procedure, is a representative method to evaluate the performance of solidified form. The contact of water, which is an abundant and generic solvent in a common environment, is considered as a beginning step and primary pathway of radioactive nuclide release and migration to the neighboring environment. Since the release behavior and mechanism of radioactive nuclides are different, the release of radioactive nuclide in the solidified form is different for each radioactive nuclide. The ANSI/ANS-16.1 is a rapid and economical method to evaluate the performance of solidified form. Although the leachability index cannot explain the complex and specified releasing phenomena, it offers the rapid and reliable estimation of them. Recently, the working group ANS-16.1 of the Standards Committee of the American Nuclear Society approved the revised procedure in 2019. The primary changes in the revised standard are constant 1-day leachant replacement interval and quantification of uncertainty using statistical method.

In this paper, the comparison of previous and current standard procedure and obtained leachability index from glass solidified form will be discussed.

Keywords: WAC, ANSI/ANS-16.1, Solidification, Leachability index

A Study on Vitrification of Glass Fiber From Nuclear Power Plant

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Various heating apparatus are used in nuclear power plant (NPP), such as pipe, steam generator, reactor, etc. The insulation materials cover the surface of them and offer safe operation condition. The insulation composed of glass fiber, coating and binding materials, etc. Lots of layers, composed of glass fibers, constructs the insulation structure. The outer surface of the insulation is coated with polymers and immobilizes them.

It is generally known that the polymeric materials are deteriorated as they are exposed to external stress, such as heat, oxidation condition, etc. And they usually result in hardened surface and separated.

In this study, the vitrification of insulation composed of glass fiber is systematically studied. The suitable composition, including additive, is invested using computer program. The fabricated vitrified form was characterized considering waste acceptance criteria (WAC) of Korea. The vitrified form of insulation satisfies the WAC and offers good operation condition of vitrification facility.

Keywords: Glass fiber, Vitrification, Insulation, Glass composition

Immobilization of Filter Waste Containing Radioactive I Using Te-Ag-Bi Glass Matrix

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During the treatment of spent nuclear fuel, radioactive iodine (^{129}I and ^{131}I) is generated in a liquefied or gaseous form in a specific process. In the case of I-131 (half-life $t_{1/2} = 8.04$ d), although the radioactivity is high, the half-life is short, and the radioactivity becomes practically negligible within 2 years, so it is not a problem in a long-term perspective. However, in the case of I-129, it is a long-lived nuclide with a very long half-life ($t_{1/2} = 1.6 \times 10^7$ y) and also has high mobility in groundwater under the condition of a repository. Therefore, despite showing a low radioactivity, research on the management of radioactive iodine from a long-term perspective has been studied. In previous study for I immobilization, borosilicate glass has been studied as I wasteform, but waste loading of I in borosilicate glass is very low (1wt% waste loading). Additionally volatility during immobilization process is high (retention ratio : 20%). In the case of ceramic type I wasteform using hot isostatic press (HIP) have been mainly studied in the United States. Although I exists in the solidified body without volatilization, it is difficult to treat a large amount of waste as a process of sintering under a high-temperature and high-pressure atmosphere and it is not suitable for handling radioactive materials in terms of stability of the process. Therefore, in this project, research was conducted to develop a new medium that can efficiently retain I in the solidified body only with a simple heat treatment process.

In the pyroprocessing, zeolite 13-X absorbent has been applied to I capturing filter in off-gas treatment process. In current study, our group uses Te-Ag-Bi composition for immobilization of I waste filter. For preparing surrogate I waste filter, Ag exchanged 13X-zeolite absorbing inactive I was synthesized (AgI loading of Ag-13X zeolite : ~54wt%, Ag (25wt%), I (29wt%)). I wasteform using TAB matrix to immobilize a I waste filter was successfully prepared without volatilization of I. Waste loading of I waste filter in wasteform was 25wt%, and 7.25wt% of I was contained in Te-Ag-Bi wasteform.

Keywords: I wasteform, Te-Ag-Bi, Ag-13X zeolite filter, I-131, Long-lived nuclide

Numerical Simulation of Spent Nuclear Fuel Transport Casks for Strain-Based Safety Evaluations

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Generally, stress-based criteria have been used to determine structural safety of SNF transport cask. However, such criteria have limitations in representing the structural behavior of cask during non-linear analysis. Because of this reason, ASME Section III, Nonmandatory Appendix FF suggested strain-based design criteria under level D service loads in 2015 and it used equivalent plastic strain. ASME Section III mentioned that since the driving mechanism for plastic distortion is the transformation of supplied energy into plastic work, the equivalent plastic strain is a better indication of the material condition than stress determination (ASME Section III). Because of these reasons, it is useful in analyzing the results after dynamic impact. This paper comparatively evaluates stress-based and strain-based acceptance criteria that are suggested in ASME Section III. Horizontal and vertical accidents drop (9 m drop) tests were performed. Finite Element Analysis (FEA) was performed to evaluate accidental drop events which were specified in NUREG-1536. Basket and canister which contain SNF were evaluated during events, and structural safety was investigated for stress-based criteria and strain-based criteria by comparing FEA results with design acceptance results. In all cases, basket and canister predominantly excess yield strain, however, did not reach acceptance strain. Compare with horizontal drop and vertical drop, horizontal drop has the highest value in basket, and vertical drop has the maximum value in canister. In strain-based design criteria, the highest maximum safety ratio was 0.27 and there had 73% of margin to reach acceptance strain.

Acknowledgements

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Keywords: Accident conditions of transport, Transport cask, Drop accident, Strain-based safety evaluation

Method for Surface Dose Measurement of Waste Filters of Primary Cooling Water System of Nuclear Power Plant

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Currently, domestic nuclear power plants temporarily store waste filters used for radioactive reduction and purification of the primary cooling water system in an untreated state. Accurate surface dose measurement is required to dispose of this waste filter. Since most of the waste filters are high doses, a method for remote measurement was derived.

First, a case of measuring the surface dose of the waste filter was investigated. The Brazilian IPEN Laboratory compared and analyzed the measured and simulated calculations to evaluate the radionuclides inventory of waste filters, and the difference in surface dose measurements according to 17 location points of waste filters was about 9.5%, indicating that the overall nuclear inventory distribution was uniform. In addition, the surface dose of radioactive waste drums at domestic nuclear power plants is measured at a total of 12 points, 3 points in the height direction and 4 points in the circumferential direction.

Considering these two cases, the diameter of the waste filter is very small compared to the radioactive waste drum, so it was determined that the distribution of nuclear mass would change only in the height direction. Therefore, the surface dose measurement of the waste filter is planned to measure three points in the height direction and apply this average value as a representative value.

Keywords: Radioactive waste filter, Surface dose, Measurement, Disposal

Improvement of Safety Design and Systems for Solar Evaporation Facility in KAERI

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The Solar Evaporation Facility placed in north site of KAERI is to treat Very Low Level Liquid Radioactive Waste (VLAW). The principle of this system is evaporation and concentration by solar energy. This facility was built in 1988 and operated until early of 2020, however, Jan. 2020. The leakage of Cs-137 was founded from environmental radiological surveillance and assessment on the 1/4 quarter. After this accident, this facility has shut down until today. The Nuclear Safety and Security Commission has ordered that Solar Evaporation Facility has to be shut down from Jan. 31th. 2020.

The most cause of this accident was difference of initial design and lack of safety system to prevent leakage even after accidents. Originally, this facility had to be designed as a non-release facility. That is to say, closed cycle. However, flooded VLAW was collected to floor drian tank and release to outside the facility after built of this facility. This facility does not have any surveillance system to check leakage of radioactive material to circumstance.

KAERI will raise the bar on safety of this facility with safety improvement works. In this study, we studied safety problems of original facility and how we will increase the safety with new facility design and systems.

Keywords: Kaeri site, Solar evaporation facility, Safety design, Facility improvement, Safety system

Investigation of the Effect of Organic Complexing Agents on the Mobility of Tc-99 in the Natural and Engineered Rock/water Interface

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The radionuclide can be released from a waste repository due to an accident, inaccurate maintenance, or leaching throughout the period of time and can leach into the soil and groundwater, which eventually will affect the environment. The migration behavior of radionuclides depends on the aqueous solution properties (pH, temperature, complexing ligands, and initial radionuclide concentrations) and sorbent properties (mineral composition, surface charge, surface area, and type of surface functional groups). This work investigated the mobility of radionuclides in waste repositories located in Gyeongju, South Korea. The adsorption behavior of Tc-99 over the natural and engineered rock materials (cement, granite, and sedimentary rock) was studied in batch sorption experiments. The adsorption experiments were conducted at a range of pHs (7, 9, and 11) and temperatures (20–40°C). In addition, the effects of the presence of organic complexing agents such as ethylenediaminetetraacetic acid, nitrilotriacetic acid, and isosaccharinic acid were investigated. The Tc-99 showed quite low K_d values representing the high mobility of Tc-99 in all the soil samples at the alkaline pH conditions. The K_d values were also significantly reduced in presence of organic complexing agents at high concentrations. As sorption is very important to reduce the leaching of radionuclides into groundwater, retarding their extensive migration to the accessible environment, Tc-99 will be a matter of concern in operating the radionuclide waste repository.

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Keywords: Technetium, Organic complexing agents, Mobility, Distribution coefficient

Status of the Revision of Regulatory Guideline for Clearance

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Since the long-term operation of facilities and equipment, the demand for clearance is gradually increasing in the domestic nuclear industry. The clearance is carried out in accordance with Article 70 of Nuclear Safety Act, Article 107 of Enforcement Decree of the Nuclear Safety Act, Articles 94 and 96 of Enforcement Regulation of the Nuclear Safety Act and Notices of the Nuclear Safety and Security Commission waste.16. Also, KINS Regulatory guideline 12.8 provides technical guidance for satisfying the relevant requirements for regulations. However, the Regulatory guideline is not a substitute for the regulations, and compliance with it is not required.

IAEA is in the process of revising Safety Guide RS-G-1.7 published in 2004, to comply the concept of IAEA Safety Standards Series No. GSR Part 3, for which purpose the Safety Guide (DS500) has been developed to share national experience and discuss existing practices of Member States. In this study, the Regulatory guideline 12.8 was reviewed and the draft guideline was developed to reflect the domestic clearance experiences and some of DS500.

The licensees have notified the plan of the clearance by taking practically representative sample of 1 kg per 200 kg of the material to be cleared based on Regulatory guideline 12.8. The sampling and analysis techniques should be selected and applied in accordance with the graded approach, and this has been reflected in the draft Regulatory guideline.

Also, the draft provides the regulatory position on the application of the theoretical method based on the clearance of surface contaminated material according to the characteristics of the material and other aspects in the draft. The licensees can assess the radioactivity concentration through the theoretical method based on the surface specific contamination by demonstrating the basis and method of applying. However, the draft doesn't provide surface-specific clearance levels. Also, when performing clearance measurements, the self-absorption needs to be taken into account in the calibration process to prevent underestimation. So, it is recommended to consider the uncertainty related to the self-absorption in the draft guideline. It is possible to apply the correction factor derived from the theoretical calculation using computational simulation code for the geometries of the material to be cleared for measurement.

IAEA continues to develop the safety document hierarchy for clearance and we will review this and continue to clean up the regulatory regime of clearance reflecting the regulatory experience.

Keywords: Clearance, Safety standard, Regulatory guideline, Safety guide

Transport of ^{238}U and ^{129}I Through Engineered and Natural Barriers: Effect of Organic Complexing Agents

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Radionuclides including ^{238}U and ^{129}I can be released from a radioactive waste repository through engineered barriers (ordinary portland cement) and natural barriers (sedimentary rock and granite) due to deteriorated facilities or unexpected accidents. Complexing agents used at the nuclear sites such as ethylenediaminetetraacetic acid and nitrilotriacetic acid are present in the waste and isosaccharinic acid will be generated in situ through the degradation of cellulose waste components. The complexing agents might affect transporting behavior of radionuclides. In this study, three barrier samples (ordinary portland cement, sedimentary rock, and granite) were investigated for the transport behavior of ^{238}U and ^{129}I in groundwater. The rock samples were characterized by XRD, XRF, FE-SEM, BET, and Zeta-potential analyzer. The sorption distribution coefficient (K_d) of two target radionuclides was determined in the presence and absence of three organic complexing agents and the effects of pH and temperature were also studied in batch adsorption systems. The K_d values of ^{238}U decreased with an increase in organic complexing agent concentrations. On the contrary, those of ^{129}I gave a negligible change under various conditions. Therefore, the above-mentioned results indicate that mobility of ^{238}U could be significantly affected by the organic complexing agents, enhancing radionuclide transport to the geosphere.

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Keywords: Uranium, Iodine, Transport, Sorption distribution coefficient, Complexing agents

Improving m-trek® (Radioactive Waste Tracking System) Using RFID and Beacon

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The Internet of Things (IoT) is now being utilized as an automation service that improves work efficiency and data quality throughout the industry. MIRAE-EN Co., Ltd. intends to acquire efficient real-time data by applying IoT technology to the basic functional set of m-trek® v1.0 (radioactive waste tracking management system). By using this, we intend to optimize the radioactive waste management and operation process.

MIRAE-EN Co., Ltd., extended the m-trek® function by utilizing RFID and Beacon sensors for waste drum tracking, warehousing/shipping, and inventory management, and performed applicability tests.

Beacon is a short-range wireless communication device based on the Bluetooth 4.0 (BLE) protocol. It can communicate with devices within a maximum distance of 70 m, and using this, location and direction information can be checked. By using this, the location of the waste drum can be checked in the space for the storage or waste treatment process, and it can be used in the radioactive waste management process. Through RFID, the function of automatically uploading the log in and out of the warehouse was implemented, and the location of the waste drum in the storage or the space for the treatment process could be checked through the Beacon sensor.

It continuously converges m-trek® with the characteristics of long-term data management and IoT technology. In addition, we plan to build an integrated radioactive waste data platform to maintain the quality of data and metadata related to radioactive waste. Such a platform will form the basis of a data science platform for future data modeling, machine learning, etc.

Keywords: Radioactive waste tracking system, RFID, Beacon, Storage, Treatment, m-trek®

Long-Term ^{60}Co Interactions and Sorption to the Cementitious and Natural Rock Barriers: Influence of Complexing Agents

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The radionuclides stored in the repository tend to release over a certain period of time due to leakage and other unavoidable circumstances and leach into the groundwater. During transportation with groundwater, the radionuclides interact with cementitious and natural rock barriers and might sorb or diffuse through those matrices. This study investigates the sorption behavior of ^{60}Co in a batch system. The cement, granite, and sedimentary rock samples and groundwater were collected near a radioactive waste repository at Gyeongju in South Korea. The rock samples were crushed into powder (from 53 to 150 microns) and pre-equilibrated with the groundwater before the sorption studies. The samples were characterized using XRD, XRF, SEM-EDS, BET, and Zeta potential analysis and ^{60}Co concentration was detected using a gamma analyzer. The sorption behavior of the selected radionuclide was described using the K_d (distribution coefficient) values considering the effect of temperature, pH (from 7 to 13), and organic complexing agents (ethylenediaminetetraacetic acid, nitrilotriacetic acid, and isosaccharinic acid). The K_d values for ^{60}Co had no significant effect of temperature. The K_d values for all the rock matrices increased linearly with an increase in pH. In case of cementitious matrix, the K_d value was three times higher for pH 13 as compared to pH 7, while in case of granite and sedimentary rock the K_d value was two times higher for pH 13 as compared to pH 7. Due to the presence of organic complexing agents at higher concentrations such as 10^{-2} M, it could be observed that the sorption of ^{60}Co onto all the rock matrices reduced significantly while their presence at lower concentrations showed no significant effect. Therefore, it could be anticipated from the above results that the lowering of sorption at higher concentrations of organic complexing agents might increase the mobility of radionuclides in the groundwater, thus leading to the transport of radionuclides to the near field sites of the repository.

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Keywords: Radionuclides, Sorption, Distribution coefficient, Cementitious matrix, Natural rock matrix

The Survey of Clearance Standard and R&D Proposal for Reuse and Recycling

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Considering the distribution according to the type of waste and the level of radioactivity, 60–70% of the waste is composed of metal (scrap) and concrete (debris) with very low or clearance levels. Therefore, research and development of clearance standards is attracting attention internationally as a preliminary step for reuse and recycling of intrinsic materials. Also, assuming that the decommissioning of the nuclear facilities is proceeded clearly in a way of cold-to-hot, it will first deal with clearance and very low-level waste, and then process LILW and HLW such as spent nuclear fuel. In addition, it can be seen as one of the efforts to reduce the amount of radioactive waste considering the capacity of the repository.

Therefore, in this paper, the overseas clearance standard system and trends are reviewed, and the priorities of domestic standardization and research and development (R&D) proposals are made.

Keywords: Clearance, Reuse, Recycle, Decommissioning and decontamination, Rubble, Scrap

Development Strategy for Incineration Treatment Facilities for Radioactive Organic Waste : Requirements and Technology

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With the operation of 25 nuclear reactors and the increase in the number of Isotope-using organization using RI, the amount of radioactive organic waste is steadily increasing every year. This organic waste is a flammable and explosive organic solvent and has potential accidental risks such as spontaneous waste ignition and fire. A stable treatment plan for the generated liquid radioactive waste is needed, however, the Korea Hydro & Nuclear Power Co., Ltd., (KHNP) and Korea Radioactive Waste Agency (KORAD) has been stored for a long time at the site until an appropriate alternative is prepared. In Korea, no incinerator for RI organic waste treatment has been developed, and there are no specific requirements for the manufacture of incineration facilities. In this paper, by reviewing the operating status of overseas incinerators and their requirements, the technical standards and requirements are proposed for the domestic development of incineration treatment technology for organic waste generated from nuclear power plants and industries.

Keywords: RI organic waste, Incinerator, ¹⁴C scrubber, Heat exchanger, Dioxin, Reactor

Development of KHNP Vitrification Technology

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Vitrification is to vitrify and incorporate radioactive waste into the stable glass matrix. Thus, release of radioactive nuclides to the environment at the disposal site can be fundamentally reduced and prevented. If this vitrification technology is applied to the treatment of combustible radioactive waste, its volume would be reduced to less than 1/20 of the original volume before treatment. In addition, the radioactive waste stability could be enhanced by a factor of 10 compared with the wastes treated by conventional methods. Consequently, safety of the final disposal of radioactive waste will be greatly improved.

KHNP has developed technology for treating LILW generated from Nuclear Power Plant. The CCIM technology has been developed for treating the combustible wastes such as dry active waste, ion exchange resin, zeolite, sludge, asbestos and so on. KHNP has conducted the feasibility and demonstration study to produce the design data of commercial plant from 1994 to 2002. Over 100 times demonstration tests were conducted using this vitrification pilot plant. Since 2009, Hanul Vitrification Facility has been successfully operated up to now. About 400 drums are converted into 13 drums of glass forms. The volume reduction factor is about 30. Based on the commercial operation experience of commercial facility, KHNP developed the large scale CCIM, and conducted the performance tests and waste treatment test.

Keywords: Vitrification, CCIM, Thermal treatment technology, LILW

Improvement of Off-site Dose Calculation Program

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In accordance with the notification of the Nuclear Safety and Security Commission (NSSC), environmental impact assessments around nuclear power plants are conducted annually and the results are disclosed to the public. KHNP developed the K-DOSE 60 program in 2012, which reflects ICRP 60, to conduct resident dose assessment. It is possible to perform population dose assessment from gaseous and liquid effluents from nuclear power plant. The current program is developed as stand alone type, so the assessment result management function is insufficient, and it is impossible to compare with other nuclear plant site results.

KHNP plans to improve the currently used K-DOSE60 program to network based operation program through R&D project and reflect various functions for user convenience. In addition, the optimal dose assessment method will be reviewed by reflecting current regulatory issues for dose assessment for multiple units.

Keywords: Off-site dose calculation, Liquid effluent, Gaseous effluent, K-DOSE60

Analysis of the Quantity of Radioactive Waste Transported From the Decommissioning of Kori Unit 1

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Currently, decommissioning of Kori Unit 1 is under preparation in Korea, and the decommissioning of commercial nuclear power plants (NPPs), which expire designed life, is scheduled to proceed sequentially. Accordingly, various types, shapes and sizes of radioactive waste generated during the dismantling of Kori Unit 1 are to be loaded into packaging containers and transport containers, which are under development, and be transported to the low and intermediate level radioactive waste disposal facility in Gyeongju.

In this study, the radioactive waste generated during the decommissioning of Kori Unit 1 is classified by radioactive waste level and type. The classified radioactive waste would be loaded into packaging containers and transport containers, which the containers under development, and the amount of the packaging containers and transport containers transported to the low and intermediate level radioactive waste disposal facility was analyzed.

Based on the transportation volume analysis, the number of trips required by each scenario (On-road transportation and Marine transportation) was preliminarily analyzed.

The estimated quantity of the decommissioning radioactive waste from Kori Unit 1 is based on the waste volume (disposal volume after reduction) for each radioactive waste level and type summarized in the "Evaluation of Decommissioning Source Terms and Wastes for the Pressurized Water Nuclear Power Plant and Heavy Water Nuclear Power Plant Final Report" issued in 2016 by Korea Hydro & Nuclear Power Co., Ltd. Through the density of each referenced radioactive waste and the volume of each type of the packaging containers, the amount of the packaging containers required is analyzed considering the maximum loading weight of packaging and filling rate of 85%, the waste acceptance criteria of the disposal facility. The packaging containers filled with radioactive wastes are to be loaded into transport containers and transported to the low and intermediate level radioactive waste disposal facility in Gyeongju through on-road transportation and marine transportation.

The number of trips for land transportations of radioactive waste is analyzed in consideration of the relevant legal requirements for transport vehicles (vehicle operation restrictions, etc.). The number of trips for marine transportation is analyzed by considering the allowable loading weight (950 Ton) of CHEONG JEONG NURI, the marine transportation vessel for radioactive waste.

Based on the analyzed number of on-road transportation and marine transportation, the estimated cost for each transportation would be evaluated, and an appropriate transportation scenario (On-road transportation, Marine transportation, or both On-road transportation and Marine transportation) would be derived based on the estimated transportation cost.

Keywords: Decommissioning radioactive waste, Radioactive Waste transport, Package container, Transport container, On-road transportation, Marine transportation

Effects of Ca or Sr Addition on Apatite for Radioactive Iodine Immobilization

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¹²⁹I is a long half-life (1.57×10^7 y) nuclide, which has been released into the environment from fuel reprocessing process. It should be managed carefully because of its high diffusivity into the environment. Since iodine has a long half-life, a durable matrix is required for long-term storage in the radioactive waste repository. The radioactive iodine generated from the reprocessing facility is collected by wet scrubbing or Ag-solid sorbent. In the wet scrubbing, the iodine is converted into the form of NaI, NaIO₃, NaOI while Ag-solid sorbent converts it into the form of AgI.

Since the converted iodine cannot be disposed directly, it should be immobilized into a chemically stable form. The iodine vaporizes easily above 500°C, so there is a limit to vitrification route. Many researchers have been investigating on the development of chemically stable matrices containing iodine. Apatite is one of the ceramic matrices which is able to immobilize iodine. The chemical formula of an apatite based waste form containing iodine is Pb₁₀(VO₄)₆I₂. This waste form loads 8wt% of iodine with showing acceptable chemical durability (3.98×10^{-5} g cm⁻² d⁻¹ as per MCC-5, 7-day Soxhlet leaching test). Generally, the chemical formula of apatite is A₁₀(BO₄)₆X₂, and the cations could be located at the A and B sites and an anion at the X site. Studies have been conducted to increase chemical durability or lower the sintering temperature through partial substitution. The most methods for synthesis show limitations such as high temperature and/or pressure. To overcome the limitation on the sintering temperature, the Pb₁₀(VO₄)₆I₂ apatite was prepared by means of wet process at room temperature (< 40°C).

In this study, we tried to mass-production of an apatite-based waste form by increasing the molar concentration in the wet process. Substitutions of Pb with Ca or Sr were conducted to modify the Pb₁₀(VO₄)₆I₂ structure. XRD and XRF analysis were performed to reveal the structural changes of the Pb₁₀(VO₄)₆I₂ apatite.

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Keywords: Fuel reprocessing process, Iodine, Wet process, Apatite

Development of Design of Glass Melt Discharge Applied to Vitrification Technology

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Vitrification with cold crucible induction melter (CCIM) is considered as one of promising treatment technology for radioactive wastes. Vitrification technology is a technology that can physically and chemically trap radioactive nuclides in glass matrix. Therefore, the release of radioactive nuclides to the environment can be fundamentally prevented.

The vitrification process developed by the KHNP CRI consists of loading glass, startup for melting glass with Ti-ring, feeding, vitrifying and discharging. The CCIM with 85cm diameter is a water-cooled melter in which electrical currents are directly induced in the glass melt by HFG. During the operation, the inner wall and bottom of CCIM is protected by cooling layers called the skull. The skull protects the CCIM from corrosion and high temperature, but it could block the hole of glass melt discharge device to make it difficult to discharge the glass melt.

In this study, to improve this problem, a new glass melt discharge device is being developed. It is made of INCONEL690 which is being used as vitrification melter of electrode, and has a protruding discharge hole (like the crown). It is expected that the crown-shaped discharge device will facilitate discharge by melting the surrounding skull.

The new device was evaluated in view of the integrity of discharging nozzle on operation environment (above 1,100°C on air environment) and the distribution of temperature to melt the skull.

Keywords: Cold crucible induction melter, Glass melt discharge, Crown-shaped discharge device, INCONEL690

Review on ALPS of Contaminated Water Treatment Facility in Fukushima Daiichi NPP

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After the Fukushima accident, massive radioactive wastes such as stone, wood, dry active waste, and contaminated water have been generated and stored in the outside. Among of the wastes, most of the contaminated water is seawater and underwater containing not only salt component (such as Na^+ , Mg^{2+} , Ca^{2+} and Cl^-) but also fission products (such as Cs, Sr and I). The contaminated water has been treated by Cs/Sr removal facilities (such as KURION, SARRY and AREVA). These are equipped with the reverse osmosis, desalination, evaporation, concentration, flocculation, coagulation, adsorption, and ion exchange method. In addition to these facilities, ALPS system is operated to meet the regulatory standards for discharge, and then sufficiently diluted. ALPS system is a facility of 62 radionuclide containing Cs and Sr using the co-precipitation (or precipitation) and absorption. Currently, three ALPS systems which are existing ALPS, improved ALPS and high performance ALPS have been operated. For the existing ALPS, TEPCO started its construction in June 2011 and completed it in October 2012. The existing ALPS is designed to treat $250 \text{ m}^3/\text{day}/\text{unit}$ with three units. The existing ALPS consists of 2 pre-treatment facilities and 14 absorption towers. For the improved ALPS, it has been in operation since 2014. The improved ALPS changed the adsorbent and added an adsorption tower to enhance the treatment performance. Similar to the existing ALPS, the improved ALPS is designed to process $250 \text{ m}^3/\text{day}/\text{unit}$ with three units. But it has 18 towers. For the high performance ALPS, its operation started in October 2014. The high performance ALPS is designed to treat $500 \text{ m}^3/\text{day}/\text{unit}$ with one unit. And unlike the existing ALPS and improved ALPS, which used coagulating sedimentation as a pre-treatment method, the high performance ALPS uses a filter method. The high performance ALPS consists of 4 filter tower and 20 absorption towers.

Keywords: ALPS, Fukushima Daiichi NPP, Contaminated water, Three ALPS systems

Evaluation of Applicability of 100 kW Plasma Torch Melting System for Radioactive Waste

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The 100 kW class PTM (plasma torch melting) system was designed for the treatment, volume reduction and disposal technology of radioactive wastes. As testing facility for treatment of various radioactive waste including the metal, concrete, liquid waste and insulator. It primarily is made up of melting chamber, thermal decomposition chamber, melt discharge and waste feeding system. In addition, the off-gas treatment system will be installed in accordance with the law of Air Quality Conservation Act Preservation by removing the NO_x , SO_x , CO_x and etc. of off-gas. The off-gas system is primarily composed of 2nd combustion chamber, selective non-catalytic reduction, centrifugal-bubbling, and induced-draft fan.

In this study, to evaluation the applicability of the PTM system, demonstration test was conducted using inorganic material (SiO_2 - CaO - MgO - Al_2O_3 rich ore). The plasma melting chamber is pre-heated by 2nd combustion device and plasma torch for 12 hours. The temperature inside the plasma melting chamber is approximately 1,600°C. The simulated waste was put into the melting chamber by the pusher feeding device with the throughput of 50 kg/hour. During the test, the power of plasma torch is maintained at 82 kW on the transfer mode. The demonstration test was performed without problems for 22 hours, and the total amount of simulated waste was 420 kg.

Keywords: 100 kW PTM system, Radioactive waste, Demonstration test, Inorganic material

Development of the Plasma Torch Melting Facility Modular Waste Supply Device

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KHNP CRI is developing 100 kW, and 1.5 MW plasma torch melting facilities. The plasma torch melting facility is to treat waste from decommissioning waste and nuclear power plants during operation. These facilities consist of a torch, a melting furnace, a waste supply device, an exhaust facility, and a discharge device. The types of waste to be treated can be classified into solids and liquids. For waste supply, it is necessary to develop a waste supply device suitable for the form.

The modular waste supply device maintains the basic structure of the furnace. Liquid and solid waste can be changed by exchanging the modular supply device. There are various types of solid waste supply devices such as pusher type and screw type. In order to supply various types of waste to the existing melting furnace, different types of waste supply devices should have been installed. Therefore, the shape of the melting furnace became more complex and the space needed to be used more widely. With the development of a modular waste supply device, the plasma melting facilities can be operated by exchanging and installing the waste supply device suitable for the shape of the waste.

The development of modular waste supply device is expected to enable various types of waste treatment and flexible facility operation of the plasma melting facility.

Keywords: Plasma torch, Decommissioning waste, Modular waste supply device, Facility improvement

Evaluation of Applicability for Megawatt Plasma Torch Melting Operation

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KHNP Central Research Institute is currently conducting the research on treatment and disposal of radioactive wastes using plasma torch melting technology. KHNP CRI has been piloted and constructed with MW (Megawatt) plasma torch melting system to evaluate the possibility of radioactive waste disposal. This facility expected to have throughput of up to 250 kg per hour and operate 40 weeks a year.

Plasma torch melting technology has been considered as a promising technology for volume reduction and waste disposal. According to IAEA TECDOC-1527 (2006), plasma technology can treat all types of waste, including metallic and non-metallic waste. In order to use the advantages of plasma torch melting technology, overseas nuclear power industries such as Switzerland (Zwilag, 2005), Japan (Tsuruga, 2006), Russia (Radon, 2005) and Bulgaria (Kozloduy, 2018) are operating commercially as a technology to treat radioactive wastes.

In this study, MW plasma torch melting tests were conducted to assess the applicability of simulated wastes (stainless steel, carbon steel and etc.). In addition, the current and voltage characteristics of MW plasma torch melting system due to exposure in the atmospheric environment was evaluated.

Keywords: Plasma torch, Radioactive waste, Applicability of simulated waste, Volume reduction

A Study on Polymer Concrete Integrity Evaluation Through Basic Chemical Resistance

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According to the Nuclear Safety Commission Notification No. 2021-26, “Regulations on the delivery of low and intermediate level radioactive waste,” high integrity containers (HIC) are defined as waste packaging containers that can maintain integrity for more than 300 years in the underground environment and disposal conditions of the disposal facility. The high integrity containers are verified in accordance with the test standards of the HIC Review Guidelines subject-specific report “Testing method for high integrity containers for the disposal of low and intermediate level wastes”. Mechanical stability, permeability resistance, chemical durability, and biological resistance are described as evaluation criteria.

In this study, the basic chemical resistance test was performed to evaluate the integrity of the disposal container material after the deterioration of aggregates and polymers by chemical solutions. The evaluation criteria should maintain the compressive strength of $100 \text{ kg}\cdot\text{cm}^{-2}$ or more after chemical resistance test deterioration. The basic chemical resistance test was referred to NRC NUREG/CP-0103 and ASTM C114. The test method is to understand the change of physical properties after immersing the material of the HIC container in a chemical solution for a certain period of time.

- HCl, H_2SO_4 , NaOH : Material deterioration due to acid/base solution with high corrosive effect and reactivity in the disposal container
- Citric Acid, EDTA : Material deterioration due to highly reactive chemicals contained in waste

Depending on the chemical resistance chemical solution conditions and time, if the compressive strength is maintained at 9.8 MPa ($100 \text{ kg}\cdot\text{cm}^{-2}$) or higher after the test, it is decided that the long-term integrity will not be affected.

Through this study, we plan to evaluate the material standards for HIC and use them as basic data for future approvals.

Keywords: High integrity container (HIC), Chemical resistance, Compressive strength, Microstructure

A Study on Periodic Evaluation of Vitrified Form From Vitrification Facility

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The KHNP has been operating Hanul vitrification facility (UVF) since 2009. The vitrification facility is composed of cold crucible induction melter (CCIM), feeding system, off gas treatment system, etc. The fabricated glass in the CCIM is discharged to the glass mold. The licensed target waste of UVF is dry active waste (DAW), ion exchange resin (IER), and their mixture. The resulting glasses are DG2, SG, and AG8W1, respectively.

The resulting vitrified form should satisfy the following technical requirements for the safe disposal.

- Compressive strength over 3.44 MPa
- Compressive strength over 3.44 MPa after radiation exposure of 10^6 Gy for DG2 and 10^7 Gy for SG and AG8W1
- Compressive strength over 3.44 MPa after thermal cycle test
- Compressive strength over 3.44 MPa after water immersion test
- Leachability index more than 6 for Cs, Sr, and Co

Also, the waste acceptance criteria (WAC) of KORAD requires the periodic evaluation of vitrified for every 4 years. In this paper, the periodic evaluation results of simulated vitrified form for UVF will be reported. The result indicates that the generated vitrified form from UVF satisfies the technical requirements in WAC.

Keywords: Solidification, Vitrification, WAC, Periodic verification

A Study on Importance of Value Criteria in Clearance Waste Recycling Scenario Using Fuzzy-AHP Process

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This study was conducted to derive evaluation criteria for evaluating scenarios when developing clearance waste recycling scenarios and to produce data that can help develop scenarios by evaluating the relative importance between evaluation criteria.

In order to establish evaluation criteria for clearance waste recycling scenarios, literature surveys and domestic status surveys were conducted, and among them, technology (safety), economy, national policy and regulatory requirements, and resident acceptance were selected. As sub-factors, radiation safety, availability of recycling technology, recycling economic effects, raw material substitution effects, compliance with regulatory requirements, government/political interference, recycling acceptability, and transport/treatment acceptability were selected.

The Fuzzy-AHP technique was used to derive the importance of the evaluation criteria for clearance recycling scenarios. The Fuzzy-AHP technique is one of the decision-making techniques, which is used to prioritize various decision-making factors through relative evaluation, and is suitable for making objective and rational judgments in the decision-making process. As a result of conducting a Fuzzy-AHP survey of 15 experts, the top factors were resident acceptance (0.359), national policy and regulatory requirements (0.328), technology (safety), (0.201), and economy (0.112). Important factors in sub-factors are recycling acceptability (0.242), regulatory requirements conformity (0.202), and (resident/worker) radiation safety (0.157).

As a result of the study, resident acceptability and national policy and regulatory requirements in the top factor, and recycling acceptability and conformity to regulatory requirements in the sub-factor were evaluated as the most important. In the top factors, resident acceptance is a problem related to people's universal perception of the nuclear industry, and it is difficult to improve in a short period of time, but it is a must-over task for recycling clearance waste. National policy and regulatory requirements are one of the most influential items in deriving scenarios, and are also factors that are greatly influenced by resident acceptance. Therefore, recycling businesses must check and comply with regulations and policies. Recycling acceptability, a sub-factor, includes the public's perception of final products manufactured through recycling of clearance waste, and must reassure the public by proving scientifically safe from a public health perspective. Compliance with regulatory requirements must be implemented according to relevant regulatory requirements no matter which scenario is adopted. Important factors should be actively considered when developing scenarios. In addition, it is expected that the results of this study will be significantly used not only in the development of clearance waste recycling scenarios but also in the management of radioactive waste.

Acknowledgements

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Keywords: Clearance waste, Recycling scinario, Fuzzy-AHP, Radioactive waste

A Study on the Optimized Segmentation of RPV and RVI of PWR for the Waste Generation Minimization for Decommissioning

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According to the PRIS (Power reactor information system) the aging of NPP (Nuclear Power Plants) around the world is accelerating.

Waste disposal accounts for the highest concern related to the decommissioning of NPPs. Among all the parts of NPP, RPV (Reactor Pressure Vessel) and RVI (Reactor Vessel Internals) are the key elements to be disposal due to their radioactivity. In Korea, there are no experiences for disposal of large components during decommissioning of NPPs. Therefore, the cutting and disposal method of the RPV in overseas cases need to be reviewed.

In this study, RPV were cut and implemented in a 3D virtual environment. When the RPV of Kori Unit 1 is optimally cut, the required storage container and working time are calculated, and when the actual work begins, the efficiency of work and an effective treatment method can be expected.

In the case of Germany, when dismantling the Stade NPP, the use of storage containers was differentiated according to the dose level of RPV, and KONRAD containers were used for the low level wastes.

In this study, the waste deposited into the largest container Type VI (3,200 mm, 2,000 mm, 1,700 mm) among the KONRAD containers. The height of RPV in Kori Unit 1 is 14.67 m in total, of which the core part excluding the head part and the bottom part is about 10 m. In order to satisfy 85% of the domestic waste acquisition regulations as much as possible, the waste was horizontally cut using mechanical method into a total of 14 equal parts and these parts were again divided vertically into 2–100 equal parts also using mechanical method to confirm the generation trend of waste storage containers.

When cutting the horizontally cut core part into 8 equal parts, 4 Konrad containers were needed, and the total cutting time was 267 min. When it was cut into 40 equal parts, 3 Konrad containers were required and the cutting time was 13,333 min. In order to minimize the exposure of workers, the method of dividing into 8 equal parts is the most optimal cutting method, but in order to minimize the volume, it is most suitable to use 3 containers by dividing it into 40 equal parts.

Through this simulation, selected method of cutting of RPV the optimized cutting method and disposal method can be expected on actual work.

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Keywords: RPV, Decommissioning, Disposal, Waste management, Cutting, Segmentation

Development of an Operating Manuals for Drying and Sampling Spent Resin, and Installing Lid on PC-HICs

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Polymer concrete high integrity container (PC-HIC) is developed to be used at all nuclear power plants (NPPs) in South Korea for packaging and disposal of spent resin. Thus, an operating manual for drying spent resin, sampling, and lid installation on PC-HIC is required to be developed. This paper aims to provide an operating manual for these processes involving PC-HICs.

High radiation level spent resin (HRL-SR) stored in long-term storage tanks is settled at the bottom of the tanks as time passes, so it requires fluidization by circulating water through circulating pipeline attached onto the long-term storage tanks. The fluidized spent resin, in a form of a slurry, is transferred via pipeline to PC-HIC to be dried by a spent resin drying system (SRDS) according to the conventional procedures used at NPP. The spent resin is flowed into the PC-HIC through a fillhead installed on it. Then, the SRDS dries out the free-standing water from the PC-HIC to achieve 1% of free-standing water by volume in the PC-HIC.

For sampling of spent resin to be carried out, first fillhead is removed from the PC-HIC via crane. A sampler is placed onto the PC-HIC via crane, inserting 60% by length of a sampling line of the sampler into the PC-HIC. Like a spoide, a rubber ball attached to another end of the sampling line is squeezed via motor of the sampler. As the motor releases the rubber ball, spent resin would be drawn into the sampling line, as same working principal as spoide. Then, the sampler is placed onto another empty sampling bottle and spent resin is released into the sampling bottle where it would be stirred thoroughly and sampled in a vial for radioactive nuclide inventory analysis.

PC-HIC filled with dried HRL-SR requires to be sealed remotely to prevent workers exposure. The lid is installed remotely onto the opening the PC-HIC using the automatic lid installation system (ALIS) developed specifically for the PC-HIC. ALIS would be remotely placed onto the PC-HIC via crane and camera monitor, and the lid would be installed automatically. All the operation by ALIS is controlled and monitored by its main control panel placed at a distance where workers would be safe from exposure.

In this paper, an operating manual for processing spent resin involving PC-HIC is introduced. Based on this research, a detailed manuals would be developed to be used in NPPs in future.

Keywords: Spent resin, High integrity container, Operating guideline, Procedures

Introduction of Automatic Lid Installation System for PC-HIC

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High level radiation spent resin (HLR-SR) generated from NPPs is planned to be packaged into polymer concrete high integrity containers (PC-HICs). Conventionally, lid installation for high-density polyethylene containers is done manually, but since PC-HICs would be filled with HLR-SR, manual installation would require workers to closely approach HLR-SR, increasing the exposure and installation time. Thus, automatic lid installation system (ALIS) for PC-HICs is developed to resolve these issues. In this paper, ALIS is introduced, and its operating principal is briefly addressed.

Conventional manual lid installation has two issues. First is that workers would be exposed to high level radiation to install the lid, leading to radiation safety issues. Second is that workers must wear protective gloves for manual lid installation, which dulls workers' senses, increasing work difficulty and installation time, and the workers exposure. To prevent such risks, ALIS is developed to remotely install the lid onto the PC-HIC filled with HLR-SR.

ALIS has three modes of operation, which are automatic, remote manual by main control panel (MCP), and local installation by local control panel (LCP). Automatic mode performs all installation processes automatically until preset values are met. Remote manual mode allows workers to install the lid using the MCP. Both automatic and remote manual modes require the workers to monitor all processes, site conditions, sensors, and alarms from the MCP and camera monitor. Local installation allows workers to install the lid using the LCP installed on ALIS when necessary.

To operate ALIS, it is connected to the MCP, and the lid is attached onto rotating plate of ALIS via electromagnet. Then, ALIS is remotely placed onto the PC-HIC by crane manually and grip devices are used to firmly fix ALIS onto the PC-HIC. The start position of lid is identified via camera monitor and match to start position marked on the PC-HIC. When set-up is completed, worker starts lid installation process. First, the lid is placed horizontally onto the opening of the PC-HIC and is detached from the electromagnet. Second, rotating plate rotates the lid until preset torque and displacement values are achieved, which are 49.0 kgf and 16.0 mm respectively. Then, ALIS is detached from the PC-HIC to be placed back into its storage box.

In this paper, ALIS is introduced, and its operating principal is briefly explained. Based on this research, a detailed manual is to be developed to allow ALIS to be used in all NPPs.

Keywords: Automatic lid installation system, Remote installation, Spent resin, High integrity container

Automated Radioactive Waste Management System With QR Code

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In these days, an industrial warehouse is managed with automated tracking and recognizing system. The radioactive waste storage facility in KAERI is a kind of warehouse. However, the movement of the radioactive waste drums in the facility and outside of the facility is managed by hand until now. The radioactive waste drum is radial symmetry shape therefore, it is not easy to distinguish and track each drum. In this study, we adapted QR code to distinguish and track drums. In before, it has a high risk to make an error when workers transcribe the location of each drum to a database. That is to say, low reliability.

In this system, workers take a picture of each drum with a QR code including information about the characteristics of a drum. Then system recognizes 3-dimensional location of each drum and presenting as a table with columns and rows. In addition, it makes it easier for the operators to recognize it by imaging. To check and make the direction of drums on the pallet, users can upload QR codes photo and designate the side of the pallet. Then system makes a QR code with information about directions on the pallets.

This system is the first stage of an automated radioactive waste management system to distinguish each drum with making and designating the direction of drums on pallets with QR codes.

Keywords: Radioactive waste drum, Management, QR code, Automated management system

Effect of Organic Complexing Agents on Cobalt and Strontium Solubility in the Groundwater

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The chelate formation of radionuclides with organic complexing agents such as ethylenediaminetetraacetic acid (EDTA), nitrilotriacetic acid (NTA), and isosaccharinic acid (ISA) influences on the transport behavior of radionuclides. For safety assessment of radioactive waste repositories, it is important to evaluate radionuclides' solubility in presence of organic complexing agents. In this study, both Co(II) and Sr(II) were selected as the model radionuclides to evaluate the effect of organic complexing agents on the solubility of radionuclides. Considering near field and far field environments, their solubility was determined at different pHs (7, 9, 11, and 13), temperatures (20 and 40°C), and initial concentrations of three organic complexing agents (10^{-5} , 10^{-4} , 10^{-3} , and 10^{-2} M). Both Co(II) and Sr(II) were more soluble at lower pHs and even in low concentrations of organic complexing agents, the solubility of the radionuclides increased. The effect of organic complexing agents on the solubility of Co(II) and Sr(II) was in order of NTA > EDTA > ISA and EDTA > NTA > ISA, respectively. Therefore, the mobility of the radionuclides in the groundwater could be enhanced by the presence of organic complexing agents and it will finally lead to a release of radionuclides to the far field environment from a radionuclide waste repository.

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Keywords: Co(II), Sr(II), EDTA, NTA, ISA, Solubility

Degradation of Cellulosic Materials Into isosaccharinic Acid Under Alkaline Conditions

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In a cementitious repository, cellulose-based wastes can be degraded into short-chain organic acids. Among soluble cellulose degradation intermediates, isosaccharinic acid (ISA) is an organic complexing agent that might inhibit the sorption of radionuclides on the natural and engineering barriers in a geological disposal facility. In addition, ISA might increase the mobility of radionuclides by the formation of radionuclides-ISA complexes. This study characterized the types and quantities of degradation products to be generated from different cellulosic materials: paper, wood, and cloth, under disposal conditions, especially focusing on ISA generation. The mixture of cloth, wood, and paper from the operation of nuclear reactors were selected as representative cellulosic materials, and were applied to batch degradation experiments under alkaline conditions (saturated with $\text{Ca}(\text{OH})_2$ at pH 12.4) at different temperatures (20, 40, and 60°C). After several months' degradation operation, dozens of cellulose degradation intermediates such as lactic acid, formic acid, and acetic acid were detected and, most importantly, ISA was one of the dominant degradation products. As temperature increased, both the cellulose degradation rate and the amount of ISA produced increased. The cellulose degradation experiments will be continued until the production of ISA reaches a steady state value as a long term study.

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Keywords: Cellulosic waste, Degradation products, Isosaccharinic acid, Alkaline conditions

Review of the Technical Standards for Solidification and Embedding

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In Kori unit 1, spent filters generated from the reactor coolant system and the spent fuel purification system, due to high radioactivity, have been stored in storage areas of NPPs according to the policy of post-decay treatment. To dispose of those high-radioactive spent filters, KHNP is considering a method of compressing and disposing of spent filters. When disposing of a high-radioactivity spent filter by compression, it is essential to consider whether it is embedded or not, so the requirements related to the embedding (referred to IAEA glossary) of the spent filter should be established. In this paper, to establish the requirements for embedding of spent filters, domestic and foreign embedding requirements were reviewed as follows, and based on these, embedding requirements will be established.

According to KORAD's acceptance criteria, non-homogeneous waste such a spent filter having a concentration of $74,000 \text{ Bq} \cdot \text{g}^{-1}$ of nuclides with a half-life of 20 years, or more are regulated as embedding targets, and embedding did not require a separate requirement unlike solidification. However, the NSSC regulations on the delivery of low and intermediate level radioactive waste included the embedding of spent filters in solidification. This causes a contradiction with the embedding requirements of KORAD's acceptance criteria because the requirement for embedding of the spent filter can be interpreted as having to satisfy the requirement for solidification.

In this regard, as a result of reviewing relevant domestic regulations on radioactive waste disposal, most regulations require that the inspection for the suitability of radioactive waste is to be processed based on KORAD's acceptance criteria, it was judged that the NSSC delivery regulations should not be interpreted as taking precedence over the acceptance criteria.

Meanwhile, in the United States, the requirements for waste form vary according to the class of waste, and the confusion of related interpretations was prevented by dividing it into micro-encapsulation, which corresponds to solidification in Korea, and macro-encapsulation, which corresponds to embedding. Additionally, NRC requires that all the of classes of waste should be meet the minimum requirements and structural requirements (Class B and C).

In this paper, the technical standards for solidification and embedding of radioactive waste were reviewed to secure disposal suitability for spent filters. In the future, it is necessary to establish the requirements for embedding of spent filters.

Keywords: Spent filter, Solidification, Embedding, Acceptance criteria

Development of a Process for Pyrolysis of Liquid Radioactive Waste

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Nuclear power plants generate various types of liquid radioactive waste (washing water, condensate water, etc.), and most are treated in the LRS (Radioactive Waste System). It is necessary to develop technology to stably treat liquid radioactive waste (caustic soda, organic fluorescent waste, etc.) that is difficult to process through LRS.

Plasma torch melting technology uses high-temperature heat (about 1,600°C) generated from the torch using an electric arc phenomenon such as lightning to reduce the volume and secure disposal suitability. This technology is used to reduce waste and secure disposal suitability through pyrolysis of liquid waste. The plasma torch melting facility pyrolysis chamber consists of an upper cover and a body, and a refractory material capable of withstanding 1,500°C or higher is installed inside, and the outer body is designed with water cooling.

The liquid waste input device is composed of a nozzle type, and it can supply and control quantitatively. The liquid radioactive waste is sprayed into the pyrolysis chamber through the input device and is pyrolyzed by high temperature. The organic material is thermally decomposed, and the inorganic material is moved to the lower part of the plasma torch melting furnace and melted.

These technologies are expected to be of great help in the treatment and disposal of liquid radioactive waste generated from nuclear power plants in the future.

Keywords: Plasma torch melting technology, Liquid radioactive waste, Nuclear power plant, Pyrolysis

Development of Scattering Radioactive Waste Treatment Process

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Nuclear power plants generate radioactive waste in gaseous, liquid, and solid forms during operation and maintenance. Since the mid-1990s, great efforts have been made to reduce the generation of radioactive waste by improving the radioactive waste management system and developing new technologies. Radioactive waste in the form of dust generated from nuclear power plants should be safely managed so that it does not disperse or leak into the environment.

The purpose of this study is to secure the stable treatment and disposal safety of scattering radioactive waste generated in the nuclear power plant management area.

scattering radioactive waste (concrete, asbestos, etc.) should be packaged in an exclusive container that minimizes leakage after being sorted at the place of origin to be pyrolyzed and solidified in a plasma treatment facility.

The dedicated container (material: polypropylene, volume: 100 L) is designed to fit into the existing radioactive waste drum (200 L) and is composed of a screw-type lid fastening method. Polypropylene is a material with excellent heat resistance, chemical resistance, tensile strength, and impact strength, and is widely used as a packaging container in the general industry. The radioactive waste separation, collection, and treatment process using a dedicated container can stably treat scattering radioactive waste, which was difficult to treat in the past, to ensure industrial safety, such as radioactive exposure, and is expected to be of great help in treatment and disposal.

Keywords: Scattering radioactive waste, Polypropylene, Plasma, Waste drum

The Management System of Radioactive Waste Using Augmented Reality Technique

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The management of radioactive waste is becoming a great concern, and it would result in environmental pollution and may be hazardous to human health if it is not properly managed. It is important to have an advanced waste management system to manage a variety of radioactive waste information. One of the most important steps of waste management is the disposal process of radioactive waste to the Korea Radioactive Waste Agency.

The purpose of this study is to verify the inside configuration information of the sealed radwaste drums with the naked eye without opening them by the aids of the AR technique. To verify the configuration information inside of the sealed radwaste drum without opening, at first, the radwaste should be classified according to the radwaste category table by the QR code, and later, the radwaste should be verified with the naked eye through AR technique by the QR code which is attached on. That is, when checking QR code, the information of the radwaste drum and its inside contents should appear in the form of AR at the corresponding drum location.

The generation and management of AR visual images and QR codes were fully connected to each other, and thus they could be implemented in the digital twin. This is to maximize the safe management of radwaste which is required by the nuclear regulatory body and Korea Radioactive Waste Agency (KORAD). Even though it is difficult to know the contents inside of radwaste drum after sealed, it is possible to know, in real-time, the inside contents of the drum if we use the developed results of this study. This developed technique can be usefully used in the total inspection of radwaste drums performed by the regulatory body, and can be a useful tool for the preliminary inspection in the place of origin and for the disposal suitability evaluation when delivered to KORAD disposal site.

Acknowledgements

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Keywords: Radioactive waste, Management, Augmented reality, QR code, Drum information

5분과

제염해체 (Oral)



Low-cost Non-toxic Surface Decontamination of Radioactive Metal Waste Using Choline Chloride-based Deep Eutectic Solvents

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Deep eutectic solvents (DES), which are synthesized from a hydrogen bond acceptor (HBA) and a donor (HBD), demonstrate outstanding metal oxide solubility, good biocompatibility, cheap production costs. Their advantages over aqueous solvents or molten salts utilized in current decontamination techniques allow the development of an efficient, eco-friendly, and economic decontamination process. By selecting different HBA and HBD, a DES with desired properties can be synthesized for an enhanced decontamination efficiency.

In this research, the feasibility of decontamination of stainless-steel (STS) wastes using ChCl:PtsA, a DES of choline chloride (ChCl) and p-toluenesulfonic acid monohydrate (PtsA), was examined by simulating the contaminant layer and removing it by leaching in ChCl:PtsA. The leaching of oxidized STS specimens was assisted by heating and stirring. Hydrated solvents were also examined for industrial application. First, oxide layers on STS wastes were simulated by high-temperature oxidation, and the simulant oxide film showed high resemblance with the real contaminant layer in terms of composition and structure. Second, leaching of oxidized STS specimens in 10wt%-hydrated ChCl:PtsA at 60°C with stirring at 400 rpm revealed that the oxide film was sufficiently removed within few hours, as observed in SEM-EDS and inductively coupled plasma-optical emission spectrometry (ICP-OES) results. Both SEM images and line EDS of leached specimens showed that oxide particles were removed and oxygen composition dropped significantly. While oxidized specimens were sufficiently leached in both hydrated and non-hydrated ChCl:PtsA, the final metal concentration in 10wt%-hydrated ChCl:PtsA after 26 hours was greater than non-hydrated solvent by a factor of 7. Around 200 minutes to 300 minutes, a sharp increase in the metal concentration gradients was witnessed, which suggests that the oxide film dissolution is finishing and STS base is starting to dissolve. This was confirmed by a concentration kinetics comparison with the leaching of STS base. This work proposed and confirmed that the surface decontamination of STS wastes using DES is possible, which illustrates a potential development of a decontamination process with enhanced economy, efficiency, and eco-friendliness. Optimization of operational parameters such as hydration, temperature, and agitation needs to be explored further. The treatment of liquid waste of ChCl:PtsA needs to be researched as well. The metal recovery of dissolved metals from the liquid waste using electrochemistry will be studied.

Acknowledgements

This research was supported by the National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT (NRF-2020R1A2C2013078).

Keywords: Green chemistry, ChCl:PtsA, Radioactive waste, Metal oxides, Leaching

Dismantling Device Concept for the Decommissioning of Calandria of PHWRs

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Wolsong unit 1, which is a CANDU-6 type PHWRs that had been operated for 30 years since 1983, was shutdown in 2019.

The calandria shell, tubesheet, end shield and fuel channel assemblies that of the highly activated structures in nuclear power plant of Wolsong unit 1, should be taken care of when it is dismantled from within reactor building.

The dismantling works should be done in consideration of the safety of radiation protection and the minimization of radioactive waste so that the safe dismantling works can be done in consideration of the size and shape of the calandria cuttings. Based on these prerequisites, the design requirements for the dismantling device were reviewed, and the cutting technology candidates suitable for the process and strategy were compared and selected.

The target of dismantling devices considered in this paper is the activated structures within the nuclear reactor building, including the calandria, fuel channel assemblies, the calandria vault that concrete structure surrounding the calandria, and the components installed in vault walls including the reactivity control mechanism.

In this paper, the applicable existing cutting techniques were reviewed to select the optimal cutting technique for dismantling the above structures. In consideration of the review results and the environment of Wolsong Unit 1, candidate technologies for cutting/dismantling each structure were proposed.

Keywords: Decommissioning, Calandria, In-situ, Dismantlement, Cutting

Decontamination of Neutron-Activated Concrete Waste by Dense Medium Separation

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After decommissioning of nuclear power plants, a tremendous amount of radioactive concrete wastes are generated. Among them, activated concrete wastes are generated by neutron irradiation, and their major radionuclides include ^{152}Eu , ^{154}Eu , and ^{60}Co . There are also minor radionuclides such as ^{55}Fe and ^{54}Mn , which have a relatively short half-life. Since these radionuclides have strong chemical bonds, it is very difficult to decontaminate them using conventional chemical methods. Thus, we tried to decontaminate those activated concrete wastes using sodium polytungstate (SPT) solution, which is well known as a harmless dense medium.

Radionuclides in activated concrete wastes are the elements that previously have existed as stable isotopes. In other words, if only sand particles containing a lot of stable Fe, Mn, Co, and Eu are separated, decontamination of radioactive ^{55}Fe , ^{54}Mn , ^{60}Co , ^{152}Eu , and ^{154}Eu will be easily achieved. Therefore, in this study, dense medium separation (DMS) was mainly focused on separating heavy mafic minerals from light felsic minerals because large amounts of metal elements are generally present in mafic minerals. Also, we named low-density sand (LDS) for light particles floating and high-density sand (HDS) for heavy particles sinking in the SPT solution.

In XRD quantitative analyses with Rietveld refinement, quartz and feldspar were the majority of LDS, and amphibole and mica were the majority of HDS. This result was also consistent with the results of the polarized microscope and scanning electron microscope. Furthermore, elemental mapping of LDS and HDS was carried out to qualitatively investigate the contents of the metal elements.

To optimize the DMS process, the density of the SPT solution and the range of sand particle size should be also selected to increase the efficiency. The experiment was conducted by changing the density of the SPT solution from $2.50\text{ g}\cdot\text{cm}^{-3}$ to $2.80\text{ g}\cdot\text{cm}^{-3}$ and the sand particle size from $75\text{ }\mu\text{m}$ to 2 mm . As a result, the optimized density was $2.70\text{ g}\cdot\text{cm}^{-3}$ and the optimized sand particle size range was $75\text{ }\mu\text{m}$ to $500\text{ }\mu\text{m}$, respectively. In this condition, DMS shows similar Fe, Mn, and Co removal efficiency ($\sim 60\%$) to the removal efficiency using 5M acids ($50\text{--}70\%$). However, DMS shows about three times higher ($\sim 80\%$) removal efficiency for Eu than that of using 5M acids ($20\text{--}30\%$). In addition, because the DMS approach can limit the volume of secondary acidic solution waste and radionuclides still exist in solid phases to be easily recovered, it is recommended to use for decontamination of activated concrete wastes.

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Keywords: Dense medium separation, Activated concrete waste, Felsic minerals, Mafic minerals, Cobalt, Europium

The Decontamination of the Surface Oxide Layer of Radioactively Contaminated Tank

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International Atomic Energy Agency (IAEA) forecasted the amount of decommissioning waste generation of NPP (PWR) of 900–1,300 MWe as 62,000 tons, of which pollution metals account for 56.45% of the total decommissioning waste generation. If the exempt radioactive waste standard can be satisfied through decontamination of low and intermediate radioactive waste, the amount of radioactive waste can be reduced. Radioactive metal waste draws a lot of attention in terms of resource recycling, and in the case of metal tanks, the goal should be on how to make metal ingots and recycle them. In this study, in order to safely and effectively treat various type of radioactively contaminated tanks generated during nuclear decommissioning, it uses high pressure water pump of over 1,000 bar optimized for tanks to physically decontaminate the oxide layer on the walls of the contaminated tank. For pollution metal waste from the primary system of a nuclear power plant, the oxide layer, including Fe_2O_3 , FeCr_2O_4 , and Cr_2O_3 , is up to 10 μm . The maximum corrosion rate observed in 40 years served waste water storage tank is 0.3048 $\mu\text{m}/\text{yr}$. Corrosion rate increases with service life, but even when the most conservative values are applied to tanks for 40 years, an oxide layer of up to 12.192 μm can be expected. For conservative assumptions, the corrosion of the decontamination simulated stainless steel 304 specimen is based on the analysis results of the radioactivity concentration of the oxide layer in the primary system. The radioactivity concentration of the oxide layer in the primary system had a significant value only for Co-60, and the tracking of the cobalt material and the observation of changes in the oxide layer were performed. When stainless steel 304 was exposed to water at 500°C for 1 hour, an oxide layer of 12 μm or more could be easily found through the cross-section observation of SEM. The decontamination system of a radioactively contaminated tank is established through efficient surface decontamination using a high-pressure pump with a rotary nozzle.

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Keywords: Tank decontamination, High pressure pump, Oxide layer, Surface decontamination, Decommissioning

Study on Soil Decontamination Near Kori Unit-1 Nuclear Power Plant Using Supercritical CO₂

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Since permanent shutdown of Kori Unit 1 in June 2017, Korea has entered the era of full-scale decommissioning of nuclear power plants. Considering the limited volume of radioactive waste repository, the expected radioactive waste after the decommissioning of the power plant is a problem to be solved. Among the various wastes generated in large quantities, soil waste has regional specificity and is difficult to decontaminate, making it difficult to manage.

Existing decontamination methods have a limitation in generating a large amount of secondary waste. Therefore, to solve this limitation, we used the supercritical carbon dioxide decontamination method. Supercritical carbon dioxide is excellent for use as a solvent for soil decontamination due to its high permeability. In addition, it is easy to separate the solute from the solvent by controlling the solubility through pressure control. The decontamination method using supercritical carbon dioxide can dramatically reduce the amount of secondary waste compared to the existing decontamination method through the solubility control through the pressure control.

In this study, a catechol-amine series, a type of chelating ligand, was used to remove metal ions from the soil, and Net4pFOSA was used to prevent the counter-anion effect in decontamination experiment process. Considering the solubility of these materials and the supercritical point of carbon dioxide, the decontamination experiment was conducted at 200 bar and 40°C.

An ultrasonic horn was used to increase the efficiency of the decontamination process. The ultrasonic horn accelerates the chemical reaction by generating ultrasonic waves. The decontamination process was performed for a total of 1 hour, and 3 minutes of ultrasonic wave generation and 3 minutes stabilization were repeated 10 times to stabilize the pressure vessel.

The decontamination efficiency was confirmed by applying the supercritical carbon dioxide decontamination method after artificially adsorbing metal ions to the soil.

Through decontamination experiments according to soil particle size, it was confirmed that the efficiency of soil decontamination is relatively low in small particle size soils.

In order to select the target clay minerals, the results were confirmed by measuring the XRD by sampling the soil near the ring. Representatively, kaolinite, montmorillonite, illite, and albite were identified in the soil of Kori. As a result of the decontamination test, there was a difference in decontamination efficiency due to the difference in adsorption power according to CEC.

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Keywords: Soil decontamination, Kori NPP, Supercritical carbon dioxide, Clay mineral, Ultrasonic horn

Application of Wireless Measurement System for Contaminated Soil in Decommissioning Site

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The miniaturization and usability of radiation detectors make increasingly possible to use mobile instruments to detect and monitor gamma radiations. Here, a Bluetooth-based mobile detection system for integrated interaction in a backpack was designed and implemented to smart equipment for detection of radioactive cesium on contaminated soil. The radiation measurement system is demonstrated in the form of a backpack and it is able to measure while being carried by a person in a wireless framework of integrated interaction. After setting the distance from the contaminated soil to 20, 50, and 100 mm, the measurement was performed, and it was analyzed that the relative efficiency was increased by about 10% compared to the commercial plastic scintillator (EJ-200). The result demonstrates that a smart equipment could facilitate to detect ^{137}Cs contaminated soil for the performance and in-situ measurement applicability. And, the measurement result was transmitted to a personal computer using Bluetooth as a portable system. Based on the results of this study, the possibility of a CdS/ZnS-loaded plastic scintillator as an alternative to inorganic scintillators has been demonstrated, and it was shown that backpack-typed mobile wireless systems can be useful for site characterization during and final phase of decommissioning.

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Keywords: Wireless measurement, Backpack, Plastic scintillator, Quantum dot, ^{137}Cs

Decontamination of Dissolved Concrete Liquors via Coprecipitation

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Nuclear decommissioning activities have the potential to generate significant volumes of radioactive wastes which require appropriate management strategies to ensure minimal economic and environmental impacts. Waste concrete from structures such as the reactor Bioshield, fuel fabrication plants, and fuel storage ponds contain varying levels of radionuclides. Removal of these nuclides would allow for a significant volume reduction of the wastes being sent for final disposal. A wet chemical treatment process is currently under development. As part of this process, it is proposed that secondary liquid wastes are to be treated by coagulation-flocculation-separation processes whereby radionuclides present in the feed are removed by the action of co-precipitation with a carrier phase (e.g. iron), brought on by pH control. Understanding and controlling this flocculation process is of importance for process optimization.

In this study a chemical treatment process for removal of radionuclides was adopted using a simple and effective precipitation method. Optimization of the coagulation-flocculation-separation process was determined based on Photometric Dispersion Analysis (PDA) and particle sizes, as well as final floc volumes and floc filtration rates. Major radionuclides Co-60 and Eu-152 present in the concrete were confirmed to be selectively removed from the dissolved-concrete liquor as precipitates by pH-adjustment to pH 9. It was evaluated that HFO and HAO played important roles in the co-precipitation of Eu and Co, respectively.

Keywords: Photometric dispersion analysis, Flocculation, Coagulation, Effluent treatment, Nuclear decommissioning, Radioactive concrete waste

Decontamination of Radioactive Wastes Using Underwater Microwave Plasma

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In 1978, the first nuclear power plant, the Kori-1 Nuclear power plant (NPP), started commercial operation in South Korea. Since then, 24 nuclear reactors had been operating in 2018. In general, the average lifetime of a nuclear power plant is 30–40 years. Consequentially the permanent shutdown of unit 1 of the Kori nuclear power plant was approved by South Korea's nuclear safety regulator in June 2017.

The term “decommissioning” is applied in its broadest sense to nuclear facilities and covers all administrative and technical actions associated with cessation of operation and service. This process comprised dismantling, decontamination, dismantling, disposal of waste, and site restoration. By the estimate from the IAEA waste generation when dismantling the PWR, the amount of metal waste generated can be 67% of the total waste, and by the Maine Yankee decommissioning report, the amount of soil waste generated can be 18.2% of the total waste.

The Chalk River Unidentified Deposit (CRUD) is a representative radioactive corrosion product accumulated on the metal surfaces of the NPPs, such as fuel rods, coolant pumps, and steam generators. Decontamination of CRUD is an essential task because radionuclides such as ^{60}Co (γ -emitter) can be deposited and reacted with the constituent of CRUD. Moreover, the soil contamination by radioactive isotopes problem is focused due to its amount and is also damaging people's health.

Plasma techniques are widely used, such as gas cleaning, surface cleaning, surface etching, surface activation, and carbon coatings. In the radioactive waste management field, a few studies are discussed which are associated with plasma technology. Using the microwave plasma in underwater conditions can provide reactive species like radicals without the addition of hydrogen peroxide because of its high density of high-energy electrons and reactive species.

The objectives of this study will be accomplished by achieving the following specific objectives: 1) Development of decontamination technology of cobalt ions in the decontamination solution by the microwave plasma in underwater conditions; 2) Apply the microwave plasma technique to decontamination and restoration of soils under the aqueous solution condition.

Keywords: Radioactive metal waste, Cobalt, Chalk river unidentified deposit, Soil, Microwave plasma, Decontamination

5분과

제염해체 (Poster)



Unit Cost Factor Development Methodology Based on Steam Generator Replacement Experience

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The unit cost factor (UCF) and the work breakdown structure (WBS) are necessary factors when estimating the cost of decommissioning a nuclear power plant (NPP). Among them, the UCF is a unit factor that can estimate the decommissioning cost according to the NPP structure. It consists of detailed work items including detailed work activities and work time, work difficulty factors (WDF), equipment, consumables, and man-power composition. Currently, the cost of decommissioning NPP is being estimated by applying the contents of the AIF/NESP-036 report at domestic and foreign. It consists of 100 items such as pipes, valves, tanks, etc. However, the three major structures of a NPP, the steam generator, pressurizer, and reactor vessel & internal, are not presented. Foreign cost estimation is being carried out by applying UCF based on decommissioning experiences.

Therefore, in this study, data from Kori unit 1 and Framatome were analyzed to review the steam generator (SG) replacement case. Through this, the UCF of the SG was derived, and the UCF of the SG was developed by applying the AIF/NESP-036 report format. In the case of detailed work, which is a major factor among them, when replacing the steam generator, the detailed work activity and work time required were calculated by referring to the old SG demolition process. Other factors were suggested with reference to the AIF/NESP-036 report. The WDF consists of five items. Height factor, Respiratory protection factor, Radiation/ALARA factor, Protective clothing factor, and Work break factor are to calculate the total working time by applying the same ratio to all UCFs. As for the equipment, the major cutting equipment and lifting equipment used in the replacement case were applied. In the case of man-power were suggested with reference to the AIF/NESP-036 report. Through this, the UCF table of the removal of SG is presented. This study will be a way to suggest UCFs for major structures of NPP that have not been previously presented. This will be able to overcome the limitations of cost estimation due to the absence of a UCF for major structures when estimating the cost of decommissioning NPP in the future.

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Keywords: UCF, Decommissioning, Cost estimation, NPP, Steam generator replacement

Estimation of Decommissioning Radiation Source Terms of the OASIS-STO Dry Storage Cask

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The activated radiation source terms of the OASIS-STO dry storage cask were estimated for the decommissioning and disposal. The neutrons emitted from the spent nuclear fuel interact with the structural material of the cask and the activated radionuclides are generated during the cask operation period. The neutron fluxes and the production rates of the activated radionuclides at the cask components are calculated by the neutron transport calculation and by applying the corresponding response function to the neutron flux, which were performed with the MAVRIC program of the SCALE code system using the design parameters of the OASIS-STO cask. The radionuclide concentrations of the cask components are evaluated based on 50 years of the cask operation period.

The calculation result shows that the significant radionuclides for the decommissioning and the disposal of the cask are Mn-54, Co-58, and Co-60. These are produced from Fe-54, Ni-58, and Co-59, respectively. Among these elements, iron and nickel are major chemical composition of the cask components but cobalt exists only as the impurity element. The cobalt amount used in this study was taken from the relevant reference, not the specific data of OASIS-STO cask. And, considering the note of the reference that the cobalt concentration of the samples was found to vary by more than an order of magnitude, it would be reasonable to judge that the predicted concentration of Co-60 is the only preliminary value with very large uncertainty.

The calculated radionuclide concentrations of cask components are compared with the limits for the very low level waste of the regulation. After 50 years of cask operation and no decay time, components such as fuel basket and canister shell are classified into the very low level waste and many other components such as high density shielding concrete are classified into the exempt waste. If 5 years of decay time is assumed after the cask operation period, the concentrations of Mn-54 and Co-58 decrease to far below than the limits but Co-60 concentration is still considerably higher than the limit. Thus, the radionuclide considered for decommissioning and disposal of the cask is only Co-60.

The result of this study would be used for the preliminary planning of the decommissioning and disposal of OASIS-STO cask even though the calculated values include very large uncertainty. The actual concentration of Co-60 could be taken by the direct measure at the cask decommissioning stage.

Keywords: Decommissioning, Radioactive waste, Disposal, Activation, OASIS, Cask

Study for Iodine Removal Mechanism Using CoAl and NiAl Layered Double Hydroxides Under Aqueous Condition

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Iodine released from nuclear power plant can cause a harmful effect on human and environment. To remedy this problem, layered double hydroxides (LDHs) can become a promising inorganic sorbent because LDH has inexpensive and facile synthesis method with minimizing gas generation for the underground disposal. In our previous study, CoAl and NiAl LDHs were investigated for their iodine removal capacities. However, the study of iodine removal mechanism under various pHs and spectroscopic scale was not fully understood. Thus, the main objective of this study is to evaluate the iodine removal efficiencies under various pHs and verify the iodine removal mechanism.

First LDHs were prepared by a hydrothermal reaction. The target molar ratio of M^{2+} (Co^{2+} or Ni^{2+}) to M^{3+} (Al^{3+}) was 2.5:1. Subsequently, I^- or IO_3^- sorption experiments under various pHs were conducted to evaluate the iodine removal efficiency. In addition, surface complexation modeling (SCM) and X-ray absorption spectroscopy (XAS) characterization were conducted to investigate the sorption mechanism.

In conclusion, all LDHs indicated high K_{ds} (1,885 and 17,472 $mL\ g^{-1}$ for I^- and IO_3^- sorption using CoAl LDH, respectively; 2,769 and 16,449 $mL\ g^{-1}$ for I^- and IO_3^- sorption using NiAl LDH, respectively) around neutral pH. At acid condition (pH 2.01–4.63), the CoAl LDH indicated the relatively low K_{ds} (0–261 $mL\ g^{-1}$ for I^- ; 0–6,528 $mL\ g^{-1}$ for IO_3^-) due to high concentration of NO_3^- from HNO_3 and LDH's structure collapse, while the NiAl LDH, only presence of NO_3^- was considered to cause relatively low K_{ds} (0–713 $mL\ g^{-1}$ for I^- ; 0–15,007 $mL\ g^{-1}$ for IO_3^-). At alkaline condition (pH 8.44–12.79), all LDHs also showed the relatively low K_{ds} (0–112 and 0–7,713 $mL\ g^{-1}$ for I^- and IO_3^- sorption using CoAl LDH, respectively; 0–354 and 0–5,507 $mL\ g^{-1}$ for I^- and IO_3^- sorption using NiAl LDH, respectively) because of the increased deprotonated surfaces and high concentration of OH^- from NaOH. These suggest that the iodine removal mechanism using LDHs is the ion-exchange between iodine and anions (e.g., NO_3^-) in the interlayer of the LDHs and the selectivity order ($IO_3^- > OH^- > NO_3^- > I^-$). In addition, the XAS results for all LDHs after IO_3^- sorption showed only first I-O shell with an inter-atomic distance of 1.78 Å, nearly identical for IO_3^- . Furthermore, SCM results will be used to compare to pH-dependent sorption results and verify the iodine removal mechanism.

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Keywords: LDHs, Iodine, Ion-exchange, SCM, XAS

Radioactive Paraffin Wasteform Treatment for Disposal Through Pyrolysis Decomposition

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Boric acid containing B-10 is used in nuclear reactor as a coolant and absorbing thermal neutrons generated during nuclear fission in the primary circuit. Boron containing coolant water waste is generated from maintenance, floor drain, analysis of plant system, decontamination and reactor letdown flows. There are two options for aqueous solution waste of boric acid. One is recycling and discharge through filtration, ion exchange, reverse osmosis and etc. The other is immobilization after evaporation and crystallization processes. The dry powder of boric acid waste liquid can be immobilized by cement, polymer, etc. Prior to the mid-1990s, concentrated boric acid waste was solidified with a cement matrix. In order to overcome the disadvantage of low waste loading of cement wasteform, a method of solidifying with paraffin was adopted. However, paraffin solids were insufficient to be disposed of as final waste. Paraffin is a kind of soft solidified material and has low compressive strength and poor leaching resistance. As a result, it was decided as unsuitable form for disposal. In KOREA, paraffin wasteform was adopted for boric acid waste treatment in 1990s. A large amount of paraffin wasteforms about 20,000 drum (200 l drum) were generated to treat boric acid waste and were stored in nuclear power site without disposal.

In this study, we want to obtain high-purity boric acid waste by oxidizing and decomposing solid paraffin wasteform through boric acid catalytic reaction. In this reaction, paraffin is separated in the form of various by-products, which can then be treated through a liquid waste treatment device or an exhaust gas treatment device. The proper temperature for sample decomposition during catalytic reaction was set through TGA analysis. Compositions of by-products and residues generated at each stage of the reaction could be analyzed to determine the state during the reaction. Finally, boric acid waste powder was perfectly separated from paraffin wasteform with disposable products through this pyrolysis process.

Keywords: Paraffin wasteform, Pyrolysis, Stepwise decomposition, Disposal, Boric acid waste

Influence of MnO on Molybdate Crystal Formation in Borosilicate Glass

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In most countries where spent nuclear fuel has been reprocessed, borosilicate glasses have been adopted as a matrix for incorporating high-level radioactive waste (HLW). However, borosilicate glass has a challenge of limiting waste loading due to the low solubility of some components, such as molybdate, phosphate, sulfate, and chloride, contained in HLW. Molybdenum, an abundant fission product in spent nuclear fuel, has low solubility ($\sim 1\text{mol}\%$ MoO_3) in borosilicate glass. The molybdenum-rich phase easily separates from the borosilicate glass during cooling and crystallizes into alkali or alkaline earth molybdate. The molybdate phase often contains alkali radionuclides such as ^{137}Cs and can crystallize in the water-soluble form of CsLiMoO_4 . Therefore, the solubility of MoO_3 in the borosilicate glasses can act as a limiting point in the waste loading, and studies have been actively conducted to develop a glass composition that improves the solubility of MoO_3 .

We developed a borosilicate glass composition by adding MnO to improve the solubility of MoO_3 . In order to simulate the slow cooling of the glass in the center of the canister, the glass specimens were cooled from $1,200^\circ\text{C}$ at a rate of $1^\circ\text{C}/\text{min}$. The addition of MnO in glass inhibited the crystallization of molybdate, and especially avoided the formation of alkali molybdate crystals, which are highly soluble in water. For the borosilicate glass containing MnO, the solubility of MoO_3 was $3.5\text{mol}\%$ ($\sim 7\text{wt}\%$) when quenched and $2.5\text{mol}\%$ ($\sim 5\text{wt}\%$) when cooled at a rate of $1^\circ\text{C}/\text{min}$. The fine structure of Mn and Mo ions in glasses was analyzed through X-ray absorption spectroscopy. The oxidation state of Mn was mostly +2 and Mn ions existed in the MnO_4 tetrahedral structure. Mo ions were present as MoO_4^{2-} anions and were specifically surrounded by ~ 4.1 oxygen atoms at a distance of $1.78\text{ \AA}$. In addition, MnO played a different structural role than Nd_2O_3 , which was previously known to increase the solubility of MoO_3 in the glasses. Nd^{3+} ions combined directly with MoO_4^{2-} and incorporated in CaMoO_4 crystals, whereas Mn ions did not form molybdate crystals and were present in the borosilicate glass matrix. Based on the experimental results, it was concluded that Mn ions were present in the glass network as an intermediate oxide rather than directly connected to the MoO_4^{2-} anion.

Keywords: Vitrification, Molybdenum, XAS, Glass structure, Radioactive ion sequestration

Effect of ZnO as Retardant of Magnesium Phosphate Cement to Solidify Radioactive Waste

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Magnesium potassium phosphate cements (MKPCs) are acid–base reaction cements formed by mixing acidic monopotassium phosphate (KH_2PO_4) and basic magnesia (MgO) in water. MKPCs have been considered stabilization/solidification materials for toxic and radioactive waste because of their excellent properties, e.g., fast setting, high early strength, good volume stability, and excellent chemical durability. Nevertheless, since MKPCs have a fast curing rate and a high heat of hydration, it is essential to use an appropriate retardant in order to manufacture them in a large size. Thus, we present calcined zinc oxide (ZnO) as a new type of retarder. This study investigates the hydration process and ZnO retardation of MKPC suspensions and pastes. The pH and electrical conductivity of the MKPC suspensions are measured, and the solid phases and ion concentrations in the solution are investigated to analyze the hydration products and processes. In addition, the reaction time is analyzed using the hydration temperature of MKPC pastes, and the compressive strengths and crystalline phases of the hardened cements are observed. Unlike boric acid, a conventional retarder, ZnO strongly retards the initial hydration of MKPCs. The H_3BO_3 delayed the entire hydration stage of the MKPCs, but the calcined ZnO strongly delayed the early hydration of the MKPCs with a $\text{pH} < 7$. Therefore, the ZnO retarder allows the fluid mixture to be handled easily during the mixing stage and the hardening stage is maintained quickly. At the beginning of the hydration, Zn^{2+} precipitated in a composition similar to that of $\text{Zn}_2\text{KH}(\text{PO}_4)_2 \cdot 2\text{H}_2\text{O}$, and its concentration in the suspensions was kept as low as $< 0.5 \text{ mM}$. The addition of ZnO accelerated the dehydration of $\text{MgHPO}_4 \cdot 7\text{H}_2\text{O}$ to $\text{MgHPO}_4 \cdot 3\text{H}_2\text{O}$. It was postulated that Zn^{2+} was adsorbed on the surface of the MgO to prevent hydration or destabilize the hydrated Mg^{2+} ions, i.e., $\text{Mg}(\text{H}_2\text{O})_6^{2+}$, to inhibit the dissolution of MgO .

Keywords: Magnesium potassium phosphate cement (MKPC), Retarder, Calcined zinc oxide, Cement hydration, Reaction mechanism

Application of VR System for the Design of a Hot Cell Facility

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The general nuclear hotcell can be designed by the designer's experiences and several well-known design guidelines for the hotcell. However, the current hotcell facility trend is toward various functions and even flexible configurations. Furthermore, traditional handling equipment such as MSM (Mechanical Master-slave Manipulator) and crane have been replaced with automated handling devices. In this regard, difficulties for hotcell facility design, including the design cost, have increased, and the traditional design method and strategy are forced to be modified and improved. Recently, various industries have utilized AR/VR (Augmented Reality / Virtual Reality) technologies to develop their infrastructure. If the traditional design knowledge for the hotcell design could be merged with these rising technologies, the conceptual design of the hotcell facility could be quickly developed and reviewed, which saves the cost dramatically in the entire design process. Moreover, if the sequential process with the various equipment has to be operated in the hotcell facility, discrete event simulation could also be utilized in the conceptual design progress with VR technology. This approach is helpful to check the interferences of each unit process intuitively during the operation and evaluate the balance of the material flow of the entire process. This study presented the practice of the design process of a hotcell facility integrated with a VR system and analyzed the feasibility of this new approach.

Acknowledgements

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Keywords: Hot cell, Teleoperation, Conceptual design, VR (Virtual Reality)

Development of Monitoring Technique for Real Time Worker Tracking in Decommissioning Site

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Nuclear decommissioning is technical process whereby a nuclear facility is dismantled to the point that it no longer requires measures for radiation protection. It is important to ensure worker safety in decommissioning site from radiation hazard. Accurate estimation of radiation dose exposed to worker is required to ensure worker safety. In addition, this activity must be done almost in real time to respond quickly.

There are two factors to be estimated to predict the exact radiation dose exposed to worker. One is worker location and the other is distribution of radiation dose according to time. In this study, worker-tracking technique in decommissioning site has been developed using computer vision technologies. The goal of this study is to track the movement of workers in the camera and find out what path it is moving in the coordinate space.

There are three steps to track worker's movement. First step is to detect workers in a camera based on their appearance. The pretrained YOLO (You Only Look Once version 4) model, which is one of the deep learning technology for object detection, was used to detect workers. The YOLO is a suitable model for this study because it generally shows good performance in real-time object detection. Second step is to track detected workers. Several techniques such as Kalman filter and Hungarian algorithm were used to track workers with unique IDs. In addition, deep learning technology called deepsort was also used to minimize occlusion issue. The last step is to know location of workers as spatial coordinates (x-y-z coordinates). The stereo vision camera technique has been applied to estimate worker's location. Two cameras that are positioned parallel to each other estimate 3D coordinates from two 2D images obtained by cameras. It is possible to know relative coordinates of workers from camera positions.

The developed models have shown sufficient possibility for application in real decommissioning sites. However, more study also will be needed on improving tracking performance.

Acknowledgements

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Keywords: Object detection, Worker tracking, Stereo camera, Decommissioning site, YOLO, Deepsort

Robot-controlled Remote Underwater Laser Cutting for Nuclear Decommissioning

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Laser cutting enables high-speed cutting and has little reaction force so it is advantageous for remote control. Also, the amount of the secondary waste is relatively small due to the narrow kerf width. In addition, laser cutting equipment requires no special maintenance. Thus, lasers are suitable as cutting tools for nuclear decommissioning. For these reasons, many researchers around the world have been developing technology to apply lasers to dismantling of nuclear facilities. In many cases, dismantling is carried out underwater to prevent radioactive exposure of workers. Therefore, it was necessary to develop a laser cutting technology that can be applied in an underwater environment.

In our group, underwater laser cutting research has been conducted since 2017. In the meantime, we have been performing underwater laser cutting studies with a focus on increasing cutting efficiency. As a result, we cut stainless steel plates up to 70 mm thick at a laser power of 6 kW and up to 100 mm thick at a laser power of 9 kW. However, the obtained results were carried out in a small water bath and the total cut length in each test was only ~40 mm. In addition, since the X-Y-Z stage was applied as a manipulator, the degree of freedom in the cutting direction was insufficient. For application to actual dismantling sites, a more flexible manipulator such as 6-axis robot is essential and long-distance cutting is also required.

Therefore, in this work, a robot-controlled remote underwater laser cutting study was performed. For this purpose, an industrial 6-axis robot that can be driven underwater was introduced. The movement of the cutting head by the robot and the operation of laser and assist gas were all controlled remotely. With a robot-controlled underwater cutting test, we could successfully cut stainless steel specimens over long distance without any problem. Through this work, we have proven the practical applicability of the laser cutting, and the obtained data is expected to be utilized in actual dismantling sites in the future.

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Keywords: Laser cutting, Underwater cutting, Remote cutting, Nuclear decommissioning, Nuclear dismantling

Application of a Concrete Waste Decontamination Technique for Recycling of $\text{Li}(\text{Ni}_{1/3}\text{Co}_{1/3}\text{Mn}_{1/3})\text{O}_2$

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KAERI is developing decontamination techniques for application in concrete waste to be generated during decommissioning of nuclear power plants. Recently, a chlorination technique was introduced as a new decontamination method, which could convert 90% of SrO through one step reaction and water washing. The chlorination technique employed different chlorination reaction tendencies among the constituent nuclides. Major components of the concrete waste including SiO_2 , Al_2O_3 , and Fe_2O_3 are not reactive with chlorine gas up to 1000°C, while SrO is readily converted into SrCl_2 at 800°C.

As a global movement towards environment-friendly society, electric vehicle (EV) market is skyrocketing nowadays especially in advanced countries. However, lifetime of a LIB in an EV is estimated to be 5–10 years, when capacity of a LIB reaches 70–80% of its initial value. In addition, used LIBs require careful management as they contain toxic chemicals and metals. These facts are driving forces that lead to recycling of used LIBs for environmental and economical purposes. Here, cathode materials are the most valuable materials as they cost around 50% of LIB cells.

Currently, there two ways for recovery of the cathode materials from used LIBs. One is a pyrometallurgical technique in which the cathode materials go through high temperature heating to recover transition metal ingot in the bottom of a reactor. This method is not widely used because of Li evaporation loss and requirement of additional processes for mixed metal ingot. Another technique is a wet process, where the cathode materials are dissolved in strong acid solution to completely dissolve constituent atoms. Each metal is recovered by adding reagents or controlling pH. However, this process requires subsequent purification steps owing to co-precipitation of metals and generates a lot of acid solution waste.

In this work, the chlorination technique was applied to recycle $\text{Li}(\text{Ni}_{1/3}\text{Co}_{1/3}\text{Mn}_{1/3})\text{O}_2$, which is a widely-employed cathode material in LIBs. Lithium was easily separated as LiCl by controlling chlorination reaction conditions. Highly pure Li_2CO_3 was achieved using Na_2CO_3 producing NaCl and water as by-products. Transition metals were employed for re-synthesis of $\text{Li}(\text{Ni}_{1/3}\text{Co}_{1/3}\text{Mn}_{1/3})\text{O}_2$ without further treatments. The overall process was successfully demonstrated in this work including electrochemical evaluation of the re-synthesized cathode materials.

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Keywords: Decontamination, Chlorination reaction, Lithium-ion battery, Recycling, Electrical vehicle

Experimental Demonstration of Remote Underwater Laser Cutting Technology in a Nuclear Power Plant Decommissioning Environment

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Laser cutting technology has been widely recognized as one of key powerful tools in decommissioning nuclear power plants as it enables remote cutting of high-radiation facilities of old nuclear power plants. Also, it allows us to reduce secondary waste during the decommissioning process, resulting in improved cost-effectiveness. However, its feasibility under practical conditions has not been thoroughly explored. In this study, we perform the experimental demonstration of remote underwater laser cutting technology in the environment analogous to decommissioning nuclear power plant facilities. Also, we analyze the requirements for the technology commercialization and confirm its feasibility.

To test the feasibility, a nuclear power plant dismantling-like Mock-up facilities was used. A robot and a transport device were installed in a large water tank. And, we built remote laser cutting utilities and devices. We also installed a fiber laser cutting system in the robot for laser beam pneumatic transmission cutting. For the cutting demonstration, we prepared two types of cutting samples (flat and curved samples) mimicking the internal structure of a nuclear power plant. Using the samples, we tested the underwater laser cutting under different cutting conditions given several parameters, such as laser power, cutting speed, and stand-off distance.

We confirmed that the remote underwater laser cutting has shown successful cutting performance under optimized cutting conditions. We also observed that the cutting performance was predominantly influenced by the robot's transfer accuracy as well as the shape recognition of the cutting target.

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Keywords: Underwater laser cutting technology, Nuclear power plant decommissioning, High-power fiber laser, Underwater robot

Identification of Critical Accidents During Operation of a Melting Facility for Radioactive Metal Waste

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During normal operation of a melting facility the furnace will be fed using a purpose-built shielded charging basket. Agitation of the melt will be required to ensure that melted material does not get stuck. This will be performed using a remote controlled mechanism which may be operated either automatically or manually.

Disruptions to normal operation of the furnace may be caused by different initiating events. Loss of power may be caused by disruptions in the grid electric supply, or by electrical failure in the equipment itself. When a disruption occurs, the melt may cool and lodge itself in the furnace. If this happens, with the conventional induction furnace design it is likely that the solid block will have to be cut and the refractory removed to allow for removal of the solidified melt. Due to the radiological characteristics of a waste melt, such work will be complicated.

There is a risk for explosions in or near the furnace due to vapor expansion. This could be caused e.g. by contact between the melt and liquids in the waste or from the cooling system in case of breakthrough. The consequences of a vapor explosion can partly be mitigated by designing the furnace foundation in such a way that the forces created by the accident are distributed in a controlled manner. Still this is a very dangerous situation. From experience it is known that loss of power occurs relatively frequently, on the order of a few times per year and that explosive incidents also occur, albeit much less frequently. There are also other potential accidents that could occur, such as a release of the melt after a breakthrough in the furnace. This is, however, less likely than the above. The ability to contain these situations is likely one of the main limiting factors in the operation of a facility for melting of waste.

The type of accidents associated with the operation of a melting facility for radioactive metal waste is assumed to only marginally differ from those associated with similar activities in the conventional metal casting industry or the current waste melting facility. However, the radiological consequences from a mishap or a technical failure differ widely. Three critical and at the same time possible accidents were identified: (1) activity release due to vapor explosion, (2) activity release due to ladle breakthrough, (3) consequences of failure in the hot-cell or furnace chamber not possible to remedy using remote equipment.

Keywords: Accident, Melting, Metal, Radioactive waste

The Application of Foam Decontamination Process by Using Radioactive Metal Waste From Nuclear Power Plant

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Decontamination technology is necessary to reduce radioactive component and contamination of complex equipment during decommissioning of nuclear power plants. The decontamination foam is considered to have potential application for the cleaning of radioactive contaminants from metallic walls, overhead surfaces, and complex components. In this study, we have developed foam decontamination process and the decontamination test was implemented by treating metal waste generated in nuclear power plant. Moreover, we have also verified the secondary waste solution process after decontamination process. As a result of decontamination process, the radioactivity of metal waste was decreased with high decontamination factor ($2\text{--}3 \text{ Bq}\cdot\text{g}^{-1} \rightarrow \text{N/A}$). In addition, the radioactivity was also not detected in waste solution after decontamination process. The decontamination foam process makes it possible to reduce the radioactivity of metal waste and secondary decontamination waste solution.

Keywords: Decontamination foam, Decontamination factor, Nuclear facility, Surfactant

Bioleaching of Cobalt by *Acidithiobacillus Thiooxidans* Immobilized Gel Beads

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The Republic of Korea government has decided to permanently shut-down two nuclear power plants: Unit 1 of the Kori and unit 1 of the Wolsong. During decommissioning of nuclear power plants, unexpected radionuclides, such as cobalt and cesium, can be leaked from nuclear power plants, consequently, contaminates soil and groundwater. Since the released radionuclides pose a significant threat to human health, great efforts are necessary to find ways to remediate radionuclides-contaminated soil. In this study, the feasibility of sulfur-oxidizing bacteria (SOB) immobilized gel beads to remediate cobalt-contaminated soil was examined. *Acidithiobacillus thiooxidans* as a representative SOB species was immobilized in two different materials: Sodium alginate (SA) and polyvinyl alcohol (PVA). Two different bioreactors containing *A. thiooxidans* immobilized SA and PVA gel beads were operated for 37 days to cultivate *A. thiooxidans*. Both SA and PVA gel beads showed significant decrease of pH without lag phase. The pH decreased from 5.0 (5.0) to 2.0 (1.5) in SA gel beads (in PVA gel beads), indicating the potential of bioleaching. After the cultivation, bioleaching process containing the PVA gel beads was operated to confirm the feasibility of *A. thiooxidans* immobilized PVA gel beads to remediate cobalt-contaminated soil. As a result, the concentration of cobalt in the effluent of the bioleaching process increased from 0 $\mu\text{g}\cdot\text{L}^{-1}$ to 27 $\mu\text{g}\cdot\text{L}^{-1}$. This result shows the feasibility of the proposed bioleaching process to remove cobalt from contaminated soil.

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Keywords: Bioleaching, Cobalt, *Acidithiobacillus thiooxidans*, Immobilization, Soil remediation

High-resolution Physiochemical Characterization of Aerosols From Plasma Cutting on Stainless Steel for Refining Internal Dose Calculation

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Cutting large radioactive structures into acceptable size for waste disposal is an essential process in decommissioning nuclear power plants. For cutting metallic waste, a plasma arc torch is commonly used because of its fast cutting speed. However, plasma arc cutting has the disadvantage of generating a large number of radioactive aerosols. Although workers' radiological safety needs to find out the physical and chemical properties of aerosols from cutting processes, the current industrial practices rely on a fixed average size without knowing its uncertainties. The properties include the chemical form, nuclide composition, and size distribution of aerosol.

To overcome such limitations, an innovative measurement system was designed and established to investigate the characteristics of aerosols from plasma arc cut on stainless steel in different cutting conditions by controlling cutting speed and power. The aerosol measurement system was composed of a cutting chamber (ca. 0.86 m³), an x-stage automatic cutting system using a servo motor, the plasma arc torch (powermax125, Hypertherm), and a high-resolution electrical low-pressure impactor (HR-ELPI+, Dekati). An isokinetic sampling was carefully employed to avoid sampling bias.

In this work, 10 mm thick stainless steel alloy 304 plates were cut in condition: cutting length set as 3 cm, cutting speed set as 10 mm·sec⁻¹, and plasma arc current set as 75 A. IMBA (Integrated Modules for Bioassay Analysis) was used to calculate the internal dose in four different ways: (1) 1 µm as a fixed activity median aerodynamic diameter (AMAD), (2) 5 µm as a fixed AMAD, (3) calculated AMAD from high-resolution number distribution, and (4) calculated AMAD of ⁵⁵Fe from ICP-OES analysis. The AMADs of the first two cases are an ICRP reference when actual distributions are unknown. The high-resolution number distribution of aerosol was converted into MMAD (mass median aerodynamic diameter) using the Hatch-Coate equation. Because the isotope distribution was independent of aerosol size, it was assumed that the measured MMAD values could be used as AMAD values. The effective dose coefficient using 5 µm showed the lowest value among the other calculated effective dose coefficients. In this experimental case, the internal dose by radioactive aerosol inhalation without considering the characteristic of aerosol can be assessed lower than the actual internal dose. Such results showed that evaluating potential radiation risk to workers during decommissioning requires source-term corrections by high-resolution physiochemical properties measurement. Future work includes the investigations of various combinations of cutting methods, cutting conditions, and materials as well as the development of on-site deployable fast measurement and correction methods.

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Keywords: Decommissioning, Radioactive metal waste, Radioactive aerosol, Size distribution

Analysis of the Foreign Cases of Historical Site Assessment of Nuclear Power Plant Decommissioning Site

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The decommissioning site of nuclear power plant should be investigated for residual radioactivity in order to reuse for other purposes. In this regard, the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) provides the step-by-step method for investigating residual radioactivity on the decommissioning site. Among these steps, Historical Site Assessment (HSA), which corresponds to the first step, determines the current site condition and serves as an important basis for judging whether the site satisfies the release criterion. However, in the case of Korea, the experiences of conducting HSA of decommissioning sites are insufficient. Therefore, it is necessary to establish an investigation procedure suitable for the domestic situation by analyzing foreign HSA cases. The object of this study is an analysis of the foreign cases of HSA of nuclear power plant decommissioning site.

In this study, the Yankee Rowe Nuclear Power Plant and the Connecticut Yankee Nuclear Power Plant were investigated. In the case of Yankee Rowe nuclear power plant, HSA was performed focusing on the areas that were not affected by contamination by radionuclides. Therefore, the areas expected not to be affected were proved that there was no actual effect by using statistical analysis of samples. the other areas were classified into Class 1, 2, and 3 according to the degree of contamination. At the Connecticut Yankee nuclear power plant, HSA was performed focusing on the areas that are expected to be affected by radionuclides. Therefore, the areas expected to be greatly affected were classified as Class 1 and Class 2 areas, and then the other areas with insignificant impact were classified as Class 3 areas.

As a result of the investigation of foreign HSA cases, each case was performed based on the HSA procedure of the MARSSIM. However, according to the characteristics of each nuclear power plant's location, topography, and degree of contamination, it was found that the areas that were considered preferentially when classifying the areas were different. The results of this study can be used for HSA of decommissioning site of domestic nuclear power plants in the future.

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Keywords: Nuclear power plant, Decommissioning, Historical site assessment, MARSSIM

Development of a Method for Detecting Abnormal Signals by Analyzing the Sound of an Emergency Diesel Generator Using the Artificial Intelligence (AI)

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This study is to detect abnormal signals by analyzing the sound during operation of an emergency diesel generator in a power plant using a Python program and an AI algorithm. We studied whether AI could distinguish normal sounds from other abnormal sounds by learning the normal sounds generated during operation of an emergency diesel generator.

In order to learn sound in AI, the recorded sound data (mono, .wav format) was divided into about 7 seconds long, and each divided sound was converted into an image (spectrogram, .png format) and used for learning. The sound used in this study was recorded before and after the start of two diesel generators, and sound data for 20 minutes and 2 hours were used. The CNN-Auto Encoder algorithm was used as the AI learning algorithm. Convolution Neural Network (CNN) is an AI algorithm specialized in extracting features from an image, and Auto Encoder is an algorithm that detects anomalies by encoding and decoding images.

It was confirmed that the difference between input and output for normal signals was small and the difference between input and output for abnormal signals was large if the CNN-Auto Encoder was sufficiently taught the sound during normal operation of diesel generator. A threshold is used as a criterion for judging normal sound and abnormal sound, and this value can be determined according to the results of training the CNN-Auto Encoder several times. In addition, as the abnormal sound, the sound mixed with the normal sound and the pre-start sound was used, and it was confirmed that it was distinguished from the normal sound.

In this study, anomalies were identified using sound and images, and this method can be applied to various tasks or facilities in the field of nuclear decommissioning.

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Keywords: Artificial intelligence (AI), Emergency diesel generator (EDG), Convolutional neural network (CNN), Spectrogram, AutoEncoder, Threshold

Evaluation of Underwater Fiber Laser Cutting Performance on Thick Stainless Steel Blocks

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Compared with other underwater cutting tools, laser cutting technology using an optical fiber provides a small generation of a secondary waste. This fact is coming from that laser-produced kerf is generally narrower. Furthermore, laser cutting is non-contact process with a specimen. Therefore, it offers the ease remote controllability because there exists a little reaction force during the cutting process. However, laser beam is vulnerable to its energy loss originating from the absorption by water. Therefore, bulky waterless cavity is necessary to reduce laser energy loss by water. The double nozzle and compressible air were used to create dense gas-filled cavity in the water. The air flow rates discharging from internal and external nozzle were $1,310 \text{ L} \cdot \text{min}^{-1}$ and $700 \text{ L} \cdot \text{min}^{-1}$, respectively. The throat diameter of internal and external nozzle were 3 mm and 12 mm, respectively. A fiber laser (IPG YLR-6000, $\lambda = 1,070 \text{ nm}$) beam was coupled with underwater cutting head. A high-powered beam emitted from laser was transmitted through 20 mm long process fiber (core dia.: $100\text{-}\mu\text{m}$) connected with cutting head. The collimation lens ($f = 160 \text{ mm}$) and focusing mirror ($f = 600 \text{ mm}$) placed inside the underwater cutting head focuses the laser beam. The focused laser beam is penetrated through the nozzle and directed toward stainless steel block (SUS 304) to melt. The standoff distance between nozzle tip and specimen's surface was commonly set to be 10 mm. Cutting head was moved inside water-filled tank with aid of computerized numerical control (CNC) apparatus. The cutting path was programed by G-code implanted in CNC.

In this study, underwater laser cutting performance was evaluated on stainless steel blocks with 48 mm, 59 mm, and 68.5 mm thicknesses using 6 kW fiber laser. For the 48 mm and 59 mm thick specimens, constant cutting speeds of $30 \text{ mm} \cdot \text{min}^{-1}$ and $20 \text{ mm} \cdot \text{min}^{-1}$ were applied, respectively. In particular, a slow speed of 5 or $10 \text{ mm} \cdot \text{min}^{-1}$ was applied for cutting relatively thick 68.5 mm thick specimen. The 68.5 mm thick block was cut by 40 mm in the left direction, 20 mm in the upward direction, and then 40 mm in the right direction. Complete cut was achieved which separates one block into two pieces. It indicates that air flow discharging from the double nozzle forms a sufficient air-filled cavity to cut 68.5 mm thick stainless block.

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Keywords: Underwater laser cutting, Fiber laser, Stainless steel block, Nuclear dismantling, Nuclear decommissioning

A Preliminary Study of Shielding Calculation of Shielding Box Surrounding the Surface Contamination Measuring Instrument for Use on Large Area Contamination

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This study is about a preliminary design of the shielding box surrounding the surface contamination measuring instrument. The purpose of the use of the shielding box is to prevent overestimation by radiation sources in the surrounding area. When measuring a large contaminated area, the measured radioactivity may be higher than the actual radioactivity. It is due to radiation emitted from sources in the surrounding area. If this weakness could be overcome, it will be possible to measure radioactive contamination accurately and quickly or to understand the distribution of radioactive contamination. In this paper, it is described that the preliminary calculation in the entire process by using the Monte-Carlo method.

The surface contamination measuring instrument measures alpha and beta/gamma. In this paper, gamma was considered only, which has great influence. Therefore, lead was considered as the material of the shielding box, and the applicability of iron and aluminum was also considered at the same time. And it was assumed that the radiation source exists only on a large ground surface, and there is no radiation source in the air. The code calculation compared the shielding performance by changing the thickness of the side of the shielding box. In addition, evaluation was performed according to the difference in height between the shielding box and the measuring instrument. When shielding box is protruded below bottom of the measuring instrument, incident gamma-rays may be shielded more. Finally, an optimization of the box geometry is performed for reducing weight of the shielding box. It is expected that above calculations will be basis for the actual design or probability of application in the future.

Keywords: Shielding box, Surface contamination measuring instrument, Large area, Shielding calculation

A Study on the Effect of Damage Inherent From Strong Oxidant to the Ion Exchange Resin in the Manufacturing Process of Strong Oxidant, HMnO₄ as a Decontamination Reagent on its Oxidative Chromium Decontamination Performance for NPP Chemical Decontamination

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A study on the preparation characteristics of HMnO₄, a strong oxidizing reagent for the NPP chemical decontamination, was conducted by the ion exchange method. In this study, HMnO₄ was prepared from KMnO₄ using IRN77 cation exchange resin. Breakthrough curve behavior was investigated by varying the KMnO₄ concentration and the flow rate of the ion exchange column. It was found that the resin was decomposed by the strong oxidizing reagent during the stay in the ion exchange column, which caused a decrease in ion exchange capacity and caused some of the resin decomposition products to enter the oxidizing decontamination reagent, HMnO₄. Through this, the resin decomposition behavior, decomposition products and resin degradation characteristics were investigated. In addition, the loss characteristics of HMnO₄, the oxidizing reagent, depending on the production conditions, due to IRN decomposition products were evaluated through the oxidative decontamination experiment using Cr₂O₃ pellets. Through the above experiment, conditions for preparing an oxidizing agent suitable for oxidative chemical decontamination were proposed.

Keywords: NSSS chemical decontamination, HMnO₄ Preparation by ionic exchange, Damage of resin by strong oxidant of MnO₄(-), IRN77

Development of AI Based Micro-Blast Scabbling Decommissioning Technology for Concrete Structures in Nuclear Power Plant

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Much concrete structure is needed to be decontaminated for reducing the waste quantity and protecting the workers safely against radiation exposure in decommissioning. The decontamination technologies for concrete structure decontamination have been developed but they have been faced the difficulties in concrete cutting job due to secondary waste production. To minimize workers' exposure and secondary waste, the micro-blat type scabbling technology using explosive material is developed. This paper shows the micro-blast type scabbling technology using explosive material, multi-dimensional contamination measurement and AI based mapping technology which can check the identification of the contamination status in surface of concrete structure as conceptual design basis. The micro-blast type scabbling technology will be a good method for concrete structure decontamination. Because the explosive material is well developed in Korea and the basis of robot system and mapping system is well supported by governmental R&D. The micro-blast type scabbling technology can be adopted towards Korean decontamination industry if the pilot scale system is proceeded successfully.

Acknowledgements

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Keywords: Micro-Blast, Decommissioning, Scabbling, Nuclear power plant, AI mapping

A Feasibility Study of Spent Nuclear Fuel Dry Storage Casks for Kori Unit 1,2 Decommissioning

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This study was conducted to recommend the most ideal Dry Storage Cask of Spent Nuclear Fuel (SNF) for Decommissioning Kori Nuclear Power Plants.

All the generated spent nuclear fuel in Korea are temporarily stored in the nuclear power plant site. Currently, most spent nuclear fuel is stored in a wet reservoir (storage pool), and some heavy water reactor spent nuclear fuel is stored after being transferred to a dry storage facility on the site.

The government established a roadmap for the construction of intermediate storage and permanent disposal facilities, but the implementation of the management policy has become unclear.

The decommissioning of nuclear power plants requires the transfer and disposal of spent nuclear fuel. However, there are no interim storage or permanent disposal facilities for spent nuclear fuel in Korea. In the case of developed countries, it takes too much time for these interim storage and permanent disposal facilities to be completed. Temporary dry storage is inevitable in order to start decommissioning the permanently suspended Kori Unit 1 and following decommissioning Kori Unit 2. Kori Unit 1 and 2 are stored in existing storage racks in storage pool, but Kori 3 and 4 are installed in dense storage racks, which are expected to have a different approach to dry storage. Kori Unit 1, which needs to be dismantled immediately, will need to be packed in a dry storage cask and stored inside the nuclear power plant site. Kori Unit 2 is also close to permanent shutdown and since the conditions are similar to those of Unit 1, it is expected that spent nuclear fuel should be dry-stored.

Spent nuclear fuel dry storage casks have already been introduced in several developed countries and are used for decommissioning. Several dry storage casks have been developed and technology development is underway to increase capacity in Korea. There are 4 main Dry Storage Casks to be compared: 21PWR Concrete Cask and 21PWR, 24PWR, 32PWR Metal Casks.

Finally, one of the decision-making processes, the Analytic Hierarchy Process (AHP), analyzed the preferred Spent Nuclear Fuel Dry Storage Cask from the perspective of various stakeholders.

Stakeholders were divided into 3 groups of government, public, and owner, and the study found that the most suitable Spent Nuclear Fuel Dry Storage Cask for Decommissioning Kori Nuclear Power Plants was on-site interim storage concept.

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Keywords: Decommissioning, Dry storage cask, Spent nuclear fuel, AHP

Review of Spent Fuel Pool Island (SFPI) Design Class

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In order to decommission the Nuclear Power Plant (NPP), Spent Nuclear Fuel (SNF) must be moved to the outside, but Korea does not have the SNF interim storage facility or permanent disposal site. In the case of Kori Units 1 and 2, it is possible to move to other NPPs on the site where there is room for the Spent Fuel Pool (SFP). However, at the time of decommissioning the Kori Units 3 and 4 in the mid-2020s, SFPs on the site are expected to be saturated, so countermeasures should be prepared. The realistic solution will be the Spent Fuel Pool Island (SFPI) method that enables the decommissioning of surrounding facilities while operating the SFP independently.

Existing Structure, System, and Components (SSCs) must be changed for the SFPI operation, and the appropriate design class (e.g., safety class, quality class, seismic category, etc.) must be defined. This can increase the safety and efficiency of the decommissioning NPP and reduce the decommissioning cost and period. However, there are no clear regulatory standards for SFPI design in Korea or the United States. In the case of the United States, the design class was classified through accident analysis, etc. while complying with the existing regulations related to the SNF storage facility.

In this paper, the design class standards used in NPPs currently in operation in Korea and the SFPI operation and design class classification cases of US NPPs were reviewed. The domestic design class standards are largely the same as those of the United States because the US regulations are applied *mutatis mutandis*. SFP cooling and makeup equipment and operating methods are also similar. In the case of the US NPP SFPI design class classification, the regulations were complied with, but the design class of existing SSCs was generally downgraded through accident analysis.

Based on the above review results, the SFPI design class of Kori Units 3 and 4 was studied in accordance with the current domestic regulations. In addition, cases of common changes in the US SFPI design class, applicability to the domestic NPP, and problems were studied. Finally, the direction to move forward in classifying the domestic SFPI design class was sought. This paper will be an important reference when classifying the design class by introducing the SFPI operation method to Kori Units 3 and 4.

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Keywords: Spent fuel pool island (SFPI), Design class, Safety class, Quality class, Seismic category

Improved Technology for Soil Washing Wastewater Through Sequential Process

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The representative technology to remediate the radionuclide-contaminated soil is the soil washing. In this process, a large amount of acidic wastewater is generated. Therefore, this study is in the process of removing the wastewater. We planned an overall process to distinguish between radioactive waste and non-radioactive waste. Therefore, it can reduce the amount of radioactive waste after the treatment for soil washing wastewater.

In this study, we remediated the wastewater through a three-step process to remove metal ions and cesium (Cs) ions. The experiment was conducted sequentially as follows. First of all, an inorganic coagulant (J-AF) was added to the washing solution to remove suspended matter in the wastewater and then, Cs⁺ ions were removed with bead adsorbents. Finally, cations were removed by neutralizing and precipitating with CaOH. Turbidity were determined after applying the flocculating agent and when comparing the organic coagulant with the inorganic coagulant, the inorganic coagulant was the most effective. NiFC and TiFC adsorbents were used instead of powder because it is possible to separate after Cs⁺ adsorption and sustainable for a long time even in strong acidity. When using NiFC and TiFC bead adsorbents in the experiment, the maximum Cs removal efficiency was approximately over 95% even in the extremely acidic condition (pH=0.4). Calcium hydroxide was added to neutralize to pH 7. Through this innovative process, the radioactive wastes were significantly reduced by removing radioactive suspended matter and Cs⁺ ions.

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Keywords: Cesium, Soil, Wastewater, Coagulant, Adsorption, Precipitation

Oxidative Dissolution of Chromium(III) Oxide by Persulfate

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In order to achieve the target decontamination factor during decontamination of the primary coolant system of a pressurized water reactors (PWRs), it is essential to repeatedly apply the pre-oxidative decontamination and reductive decontamination processes in consideration of the characteristics of the corrosion oxide film formed inside surface of the primary coolant system. In the pre-oxidative decontamination process, it is common to use permanganate (MnO_4^-) as a chemical decontamination agent. Typical examples include NP (nitric acid permanganate) or SP (sulfuric acid permanganate) and HP (acid permanganate) processes applied in acidic conditions and AP (alkaline permanganate) process applied in basic conditions.

In the case of repeatedly applying the pre-oxidative decontamination process using permanganate as an oxidizing agent, that is, when the second pre-oxidative decontamination step is applied after the reductive decontamination step applied following the first pre-oxidative decontamination step is applied, if a considerable amount of manganese ions are present in the decontamination solution, most of the oxidizing agent injected into the second pre-oxidative decontamination step reacts with manganese ions to be converted into manganese oxide (MnO_2) and consumed, hence significantly decreasing the efficiency of the pre-oxidative decontamination process.

Therefore, in this study, persulfate was selected as an oxidizing agent capable of replacing permanganate as one of the solutions to the above problem, and the possibility of an alternative pre-oxidative decontamination process was investigated through evaluation of the dissolution characteristics of chromium(III) oxide (Cr_2O_3) by persulfate.

Dissolution of chromium(III) oxide by persulfate was able to obtain meaningful results under basic solution conditions, and the effect of process variables such as persulfate concentration, NaOH concentration, temperature, *etc.* on the oxidative dissolution behavior of chromium(III) oxide was evaluated.

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Keywords: Dissolution, Chromium(III) oxide, Pre-oxidative decontamination, Permanganate, Persulfate

A Study on Estimation of Radiation Exposure Dose During Dismantling of Steam Generator in Decommissioning of Nuclear Power Plant

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In 2017, Kori-1 which is Korea's first commercial nuclear power plant (NPP) was permanently shut down. Because the large components were neutron activated or contaminated caused by activation and corrosion products (Chalk River Unidentified Deposit: CRUD), it is important to safely segment and dismantle the large components from the radiation exposure during decommissioning of NPPs. Among them, steam generators (SGs) are the largest components which are generated 2–4 depending on the reactor type per unit and its tasks in the dismantling process require a large number of workers. Hence it is necessary to consider the radiation exposure dose of workers which are involved in dismantling of the SGs. To ensure the safety from radiation exposure, it should be performed to estimate the radiation exposure dose of the dismantling workers and to optimize removal process. In this study, the optimized removal processes of SGs were analyzed based on NUREG/CR-1595. To estimate the effective dose for the main organs of the dismantling workers, MCNP and the ORNL-MIRD phantom were utilized. We demonstrated the feasibility of using the simulation results to estimate the optimized radiation exposure dose for the dismantling workers of SGs.

Keywords: Decommissioning of NPPs, Steam generators (SGs), Dismantling workers, Radiation exposure dose, MCNP, ORNL-MIRD phantom

A Study on Estimation of Uniformity of Ingots Using Characteristic X-rays After Induction Melting of Radioactive Waste

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To determine the radioactivity of the ingot generated from the induction melting for metal radioactive wastes, the horizontal uniformity of the radionuclides of the ingot should be estimated. However, it is difficult to estimate the horizontal uniformity of the ingot because of their high density and thick thickness in general. In this study, we estimated the uniformity of the ingot using a 2D imaging from characteristic X-rays of the elements in ingot. If the energy of the incident X-ray is higher than the binding energy of the K-shell, a K-shell characteristic X-ray with an energy is emitted. Since atoms have their own characteristic X-ray energies, the reconstructed images for each atom were able to estimate the distributions of the elements. The elements which used to estimate the uniformity of the ingot were Co and Fe, whose K_{α} X-rays were analyzable using X-ray fluorescence (XRF) device (ZSX Primus IV: Rigaku). The uniformity of each element in ingot can be calculated by using the equation of the integral uniformity. Our results indicated that the feasibility of estimating the uniformity of the ingot using reconstructed images of the characteristic X-rays was demonstrated.

Keywords: Induction melting, Radioactive waste, Horizontal uniformity of ingot, Characteristic X-ray, XRF

Decontamination of Cs-contaminated Soil by Using Non-acidic Solution

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Cs-137 is considered to be one of the main radiation exposure sources due to its high fission yield and short half-life. In general, the soil-washing method with a strong acid solution is widely utilized for the decontamination of Cs-contaminated soil. Even though its high efficiency, the acid-based soil-washing method shows unnecessary side effects such as acidification of soil, destruction of crystal structure, and dissolution of various metallic elements, inducing remarkable difficulty in further reuse of soil after the decontamination process. In the present work, optimistic condition for the decontamination of Cs-contaminated soil was investigated by using the soil-washing method with a non-acidic aqueous solution for the mitigation of various disadvantages caused by strong acid solution. The Cs-contaminated soil was synthesized by mixing 50 g of bentonite with 0.01 M of CsCl solution. The decontaminant solution was prepared with 0.1 M, 0.5 M, and 1.0 M of BaCl₂ solution, together with 2.0 M of HNO₃ for comparison. For the decontamination reaction, a liquid to solid ratio of 0.05 g·ml⁻¹ was employed at room temperature. As a result, 2.0 M of HNO₃ showed the highest removal efficiency among various decontaminants, and the Cs-bentonite samples decontaminated with BaCl₂ solution presented increasing removal capacity with decreasing BaCl₂ concentration. The duplicated experiment performed at T = 50°C revealed slightly increased removal capacity for most of the samples investigated. According to the quantification analysis result, the overall decontamination reaction reached its equilibrium state after 1 hour of reaction time, presenting the maximum removal efficiency of *ca.* 45% compared with the HNO₃ solution. The experimental data indicated that the reaction temperature and barium concentration can be crucial parameters affecting the decontamination efficiency of the BaCl₂ solution. The result obtained in the present work can be background information to compensate for the disadvantages of the conventional soil-washing method with acid solution.

Keywords: Decontamination, Cesium, Bentonite, Soil-washing

Analysis of Foreign Cases for Groundwater DCGL Derivation

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Kori Unit 1 was permanently shut down in 2017. In Korea, Final Decommissioning Plan (FDP) must be submitted to NSSC (Nuclear Safety and Security Commission) for decommissioning. The NSSC Notice No.2020-8 states that Derived Concentration Guideline Level (DCGL) should be evaluated and reported in FDP. When deriving DCGL, all media that can cause radiological effects on the public or worker should be considered. If groundwater contamination exists at the time of site reuse, and there is a possibility of using the groundwater, DCGL for groundwater must be derived. In Korea, many studies related to soil and building DCGL have been conducted, but the experiences of deriving groundwater DCGL are insufficient. Therefore, it is necessary to analyze the foreign cases of groundwater DCGL derivation in order to derive groundwater DCGL suitable for the domestic situation. In this study, the foreign cases of groundwater DCGL derivation of decommissioning nuclear power plants were analyzed.

In this study, Yankee Rowe, Connecticut Yankee, and Zion nuclear power plants were analyzed, focusing on the dose criterion, exposure scenario, and radionuclide for the derivation of groundwater DCGL. Yankee Rowe nuclear power plant applied $0.25 \text{ mSv}\cdot\text{y}^{-1}$ as a dose criterion suggested in 10 CFR 20 subpart E. The resident farmer scenario was applied to the exposure scenario, and H-3 found in groundwater was evaluated as the target radionuclide. Connecticut Yankee nuclear power plant applied $0.25 \text{ mSv}\cdot\text{y}^{-1}$ as a dose criterion. The resident farmers scenario was applied to the exposure scenario, and groundwater DCGL was derived for 20 radionuclides expected to exist in nuclear power plant site. Zion nuclear power plant applied $0.25 \text{ mSv}\cdot\text{y}^{-1}$ when deriving groundwater DCGL. Based on the resident farmer scenario, groundwater DCGL was derived for 8 radionuclides expected to exist in nuclear power plant site.

In this study, the cases of groundwater DCGL derivation in foreign decommissioning nuclear power plants were analyzed. As a result of the analysis of overseas cases, $0.25 \text{ mSv}\cdot\text{y}^{-1}$ suggested in 10 CFR 20 subpart E was applied as the dose criteria, and in general, the resident farmer scenario was applied to exposure scenario to derive groundwater DCGL. Radionuclides expected to exist at each decommissioning nuclear power plant site were selected as target radionuclide. The results of this study are expected to be used as basic data when deriving groundwater DCGL in Korea.

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Keywords: Decommissioning, Groundwater DCGL, Dose criterion, Exposure scenario, Radionuclide

Derivation of DCGL for NPP at Wolsong Site by Using RESRAD-ONSITE Code

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The Derived Concentration Guideline Level (DCGL) is the permissible residual radionuclide concentration that determines the criteria for the release of the decommissioned site. Based on the various site-specific parameters and relevant data obtained from the previously investigated case for the US nuclear facilities, further reliable and site-specific DCGL for the nuclear power plants located at the Wolsong site was preliminarily evaluated with the RESRAD-ONSITE computational code developed by Argonne National Laboratory. In the present work, a geological structure model divided into three different parts, *e.g.* contaminated zone, unsaturated zone, and saturated zone, were established according to the geological characteristics at the Wolsong site. The migration and retardation behaviors of radionuclides were predicted with the distribution coefficients (K_d) averaged from the data selectively collected from the JAEA sorption database considering the pH of the natural groundwater at the Gyeongju area. Moreover, the dietary and hydrological data were additionally employed to derive the preliminary DCGL of nuclear power plants at the Wolsong site, and the obtained DCGL values were cross-checked by comparing with the previously reported literature data. Finally, for the determination of the primary parameters remarkably influencing the DCGL result, relevant sensitivity analysis was performed for the site-specific parameters together with priority 1 parameters previously defined in the RESRAD-ONSITE code. The results obtained in the present work are expected to provide technical background information for the successful reuse of the decommissioned sites.

Keywords: DCGL, RESRAD, RESRAD-ONSITE, Decommission

A Study on Radioactive Waste Measurement Method Using NaI(Tl) Detector

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In the process of decommissioning a nuclear power plant, a large amount of radioactive waste with various shapes, materials, and densities is generated. In the case of metallic decommissioning waste, it is complicated to accurately measure the degree of contamination because the location, shape, material, density, Etc., of each contamination source are different even if they are contaminated with the same amount of radioactivity. Therefore, it can be evaluated as a different value depending on the measurement and evaluation method. This is because the density correction factor according to the magnetic absorption effect is different according to the characteristics of the metallic.

The existing radioactive waste density correction factor studies did not consider the characteristics of metallic wastes but evaluated the standardized apparent density, so it was difficult to evaluate the degree of contamination of radioactive wastes accurately. Therefore, it is necessary to apply a conservative methodology in the measurement to increase the credibility of the regulatory body and the acceptance of the public. In this study, a methodology to conservatively evaluate radioactive waste using a 3D camera, NaI(Tl) detector, and MCNP simulation is presented.

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Keywords: Radioactive waste, Decommissioning, NaI(Tl) detector, 3D Camera, MCNP simulation

Preliminary Study on Laser Cleaning for Decontamination of the Primary Circuit in Nuclear Reactors

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In nuclear decontamination, laser cleaning has many advantages. Laser cleaning is effective enough to show a decontamination factor of $\sim 100\%$, and high-speed decontamination is also possible. In addition, the greatest advantage is that it can significantly reduce the amount of hazardous secondary waste compared to other decontamination technologies. And it has high flexibility of decontamination work by utilizing optical fiber delivery of laser beam. For these reasons, the use of laser cleaning in nuclear decontamination continues to increase.

In general, industrial laser cleaners using pulsed lasers with a low average power density (sub-MW/cm²) have been mostly applied to decontamination. These laser cleaners have the advantage of being able to clean metal surfaces without damaging the base material in industrial applications. However, in the case of coolant piping of the primary circuit, radioactive contaminants are not only deposited on the surface but also some of them penetrate deep into the base metal due to the stress corrosion cracking (SCC). Therefore, it is possible to thoroughly decontaminate not only the rust and surface oxide but also the metal base by etching more than 30 μm . On the other hand, the method of instantaneous vaporization and removal by irradiating a laser beam with a high average power density (sub-GW/cm²) is suitable for in-depth decontamination because even the base metal can be etched. This is achievable using single-mode lasers with high beam quality. However, related studies have not been sufficiently conducted.

Therefore, in this work, a preliminary study on laser cleaning for decontamination of the primary circuit in nuclear reactors was conducted using a 300-W single-mode continuous-wave fiber laser. High power density was obtained through focusing in $\sim 50 \mu\text{m}$, and a surface etching test was performed using a 2-D Galvano scanner. The etching depth of the stainless steel surface was also measured while changing the process condition.

Acknowledgements

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Keywords: Laser, Laser cleaning, Laser decontamination, Nuclear decontamination

Review of Global Project Management Standards for the Decommissioning of Nuclear Facilities

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In Korea, starting with Kori Unit 1, the decommissioning of nuclear facilities is expected to be started in the near future. The decommissioning of a nuclear power plant is a long-term project, and the management for the work performed will be the key to a successful decommissioning. Therefore, in this study, we review the global project management standards and refer to the insights necessary for the decommissioning project.

First, the PMBOK guide uses a process as the basic unit for project management, and defines that project management consists of a combination of several processes. The project is carried out based on the life cycle and logically combined according to the life cycle of the unit processes, which is the minimum stage of project management, in order to carry out the specific project goal. The tasks performed in each process in the life cycle are divided into five stages: (1) Initiating: define a new project or a new phase of an existing project with permission to carry out the project, (2) Planning: determining the scope of the project, the adjustment of objectives, the direction of activities, etc., (3) Executing: performance of tasks defined in the project management plan, (4) Monitoring and controlling: track, review and control the schedule of the project and performance, (5) Closing: combination of processes required to successfully complete a project or phase.

Second, there is ISO 21500, and since this approach is fundamentally constructed based on PMBOK, the type and numbers of project lifecycles and knowledge areas are similar. The project life cycle is divided into initiation, planning, executing, monitoring and controlling, and closing. The area group consists of integration, stakeholder, scope, resource, time, cost, risk, quality, procurement, and communication.

Third, PRINCE 2 uses a different approach from PMBOK as a constructed project management method. It determines the scope of the project goal, which is defined as an acceptable value for the performance goal, and may be a key performance indicator in some organizations. It should be managed by 7 themes similar to the knowledge area of the PMBOK guide and tailored according to the characteristics of the project. The 7 themes defined in PRINCE 2 are managed based on 7 processes and are as follows: (1) Starting up, (2) Initiating, (3) Directing, (4) Controlling, (5) Managing products delivery, (6) Managing stage boundary, and (7) Closing.

Lastly, the life cycle of a project defined by the IAEA includes (1) Identification, (2) Initiating, (3) Development and definition, (4) Execution, and (5) Closeout. The big difference from PMBOK or ISO 21500 is that there is no monitoring stage and therefore no feedback to the planning stage. Here, returning to the identification stage from the closeout should be understood as the utilization of the project, not the concept of feedback. As for the areas to be managed, there are integration, scope, time, cost, quality, human resource, communication, stakeholders, risk, HSE, and lessons learned. There are also specific nuclear PM areas: radiation dose and radioactive waste management, licensing, emergency planning and response, and security and safeguards.

In summary, we think that the concept of the process stage and knowledge areas of PMBOK and ISO 21500 can be used as a configuration of the project management system when establishing the nuclear power plant decommissioning process. By applying some of the principle concepts and themes of PRINCE 2, it can be helpful in developing the principles and standard functional organization, and it will be possible to define detailed contents or activities for the project management of nuclear decommissioning by referring to the areas suggested by the IAEA.

Keywords: Decommissioning, Project management, PMBOK, PRINCE 2

Analysis of Metal Ion Concentrations for Secondary Waste Evaluation of Underwater Laser Cutting

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The generation and treatment of radioactive waste is one of the fields receiving the most attention in the nuclear power plant industry. Recently, a lot of research is in progress, especially research on reducing secondary waste in the dismantling process is one of them. Among the many methods used in the dismantling process, laser cutting is a method of reducing secondary waste.

In this study, the type and concentration of metal ions generated during the laser cutting process were measured. We try to infer the types and concentrations of secondary wastes to be determined and generated. Laser cutting was carried out in a water tank measuring 1,760 (L) × 580 (D) × 550 (H). 50, 70, and 80 mm thickness of stainless steel was cut using a fiber laser with an output of about 6 kW. The volume of water in the tank at this time was about 560 liters. The cutting time is about 5 to 15 minutes, depending on the thickness of the cut specimen. Specimens made of stainless 304, which are widely used in nuclear power plant structures, are used. During the laser cutting process, the most of melt of the stainless specimen is cooled by water and precipitates, some of which dissolves in the aqueous solution. The concentration and type of ions were analyzed. It was expected that the concentration of metal ions would increase in proportion to time and thickness in underwater cutting. In the analysis result, there was no difference in the concentration of dissolved metal ions come from stainless alloy Metal ions. the metal ion were analyzed using ICP-AES. The concentration of chromium and nickel ions, which are a problem in environmental wastewater, is lower than the concentration of environmental emission. In this study, the environmental impact and types of secondary wastes that can be generated by laser cutting are reported.

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Keywords: Laser cutting, Underwater cutting, Secondary waste, Environment analysis

Study on the Residual Radioactivities in Soil in Preparation for the Decommissioning of AGN-201K

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AGN-201K is the only educational reactor in Korea and began operation in 1982, which had been donated from US Colorado State University in 1976. Due to the characteristic of zero-power reactor, the maximum operation allowable power is very low and the maximum operating power allowed less than 10 Watts. Thus, the accumulated amounts of fission products are very small, and consequently insignificant level of radiation is contained in structure and soil. Since AGN-201K is located within the university adjacent to the main street of residential area, the geographical characteristics must be considered. The evaluation of the soil radioactivity was conducted using Resident Farmer Scenario. DCGL was evaluated from the soil activity analysis to compare site release criterion. Conservative assumptions were used because the dismantling plan was not set up.

For the evaluation, RESRAD-ONSITE computer program and hand calculations were jointly used. Input parameters were mainly taken from the design data, but some acquired input data were assumed as the default values in RESRAD-ONSITE program, or valued recommended in the regulatory guidelines of KINS/RG-N02.02.

The main premise was that the contamination of the soil surface directly under the reactor is uniform from the surface to a depth of 30 cm, and equally contaminated throughout the total ground area (287.92 m²) of AGN-201K.

DCGL_w of K-40 and Fe-59, which had relatively high concentration in soil contamination, were calculated. As K-40 was the highest contributor on the soil contamination, the dose evaluation from K-40 was separately performed to calculate the expected exposure dose rate. As a result, the annual exposure dose rate at the start of dismantling is about 0.05 mSv/yr, satisfying the regulatory limit value of 0.1 mSv/yr. Therefore, it was approved that additional restoration or decontamination are not necessary.

Despite the conservative assumptions for many uncertainties, results showed that the exposure dose rate could sufficiently satisfy the site release limit of 0.1 mSv/yr. However, in the preliminary work for full-scale dismantling, many assumptions applied in this study shall be updated as the realistic values acquired through the site survey. In addition, reevaluation shall be performed based on the site measurement data as the realistic source-term rather than tentative activation analysis results used in this study.

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Keywords: AGN-201K, Residual radioactivity, Soil activity, DCGL, RESRAD

A Review of the Feasibility of Applying the Spent Fuel Pool Island During NPP Decommissioning

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For the decommissioning of a nuclear power plant (NPP), it is essential to safely manage the spent fuel stored in the power plant. Despite the fact that the decommissioning project of NPP is being promoted in earnest in Korea, the spent fuel management plan has not been established yet. Therefore, it is highly likely that the establishment of a spent fuel management plan for the NPP decommissioning project will emerge as a pending issue in the future.

Looking at the cases of other countries that have experience in NPP decommissioning, there is a case in which spent fuel is safely stored for a certain period by applying the concept of “Spent Fuel Pool Island (SFPI)”. Mainly, it was a method that changed the design of the existing Spent Fuel Pool (SFP) so that it could be completely separated from the existing power plant facility and operated independently. Therefore, it is necessary to review these management measures for the decommissioning of NPPs in Korea.

There are several advantages to be expected of applying the SFPI concept. The entire decommissioning period of the NPP can be shortened, and the radiation exposure of workers can be reduced. In addition, it is possible to configure the cooling facility for decay heat with a low capacity, so it is expected to reduce the spent fuel management operating cost.

In the absence of a long-term spent fuel management plan, SFPI may be one of the alternatives that can be considered as a spent fuel management method for the decommissioning of NPP. In order to apply SFPI, it is first and foremost to review at what level the decay heat generated in the SFP can be properly removed when the SFPI concept is applied. This can be confirmed through the calculation result of the decay heat of the SFP considering the cooling period.

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Keywords: Decommissioning, SFP, Decay heat, SFPI

Development of Non-Routine Cost Factors in NPP Decommissioning Cost Estimation

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Nuclear Power Plant (NPP) decommissioning cost consists of activity dependent cost, period dependent cost and collateral cost. Among them, the activity dependent cost is mainly affected by plant inventory and unit factor (UF). UF, which is mainly classified into a routine factor (RF) and a non-routine factor (NRF), is developed to provide a rapid and simple method for cost estimation of dismantling large numbers of components and structures. RF is used for cost estimation of routine operation in NPP decommissioning, such as removal of pipe, pumps, and valves, etc. NRF is developed for non-routine decommissioning activities, such as removal of RPV, steam generator, and pressurizer, etc. In this paper, the relationship between NRF and equipment specifications explained in AIF NESP-036 is reviewed and NRF calculation is briefly addressed

Unlike RF of which the reference data could be obtained from both nuclear and non-nuclear related industries, NRF lacks reference data for its development as the NRF is developed for equipment, which are not routinely dismantled, such as RPV, turbine generators, and condensers, etc. Commonly, NRF calculated in AIF NESP-036, or actual cost of removal is modified to be used for cost estimation of decommissioning. NRF modification is done by comparing the equipment specifications, including size and weight, of the reference equipment with the specification of the targeted equipment. AIF NESP-036 suggests a simple method of calculating the NRF based on equipment specification, and this method is commonly used in current decommissioning cost estimation.

NRF modification based on dimension is widely accepted, a site specific NRF could be developed. NRF is developed on activity duration, crew size, and material and equipment costs. Activity duration is only partially dependent on equipment specifications. Activity duration depends on sub-activities composing the removal of an equipment. Some activities depend on the equipment specification such as the duration for segmentation or decontamination, while some other activities are not affected by equipment specification, such as work platform set-up or crane set-up. Furthermore, based on the equipment specification, site-specific conditions or decommissioning strategy, sub-activity composition and crew size may be changed.

Thus, it is important to consider sub-activity compositions, crew size and activity duration when NRF is to be developed or modified. From this method, more accurate result could be obtained, but in the purpose of quick and simple decommissioning cost estimation, NRF modification developed by AIF NESP-036 is widely acceptable.

Keywords: NPP decommissioning, Unit factor, Non-Routine factor, Decommissioning cost estimation, AIF NESP-036

Preliminary Transportation Risk Assessment for Kori Unit 1 Decommissioning Wastes

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Kori Unit 1, the first commercial nuclear power plant (NPP) in Korea, was permanently shut down in June 2017, and full-fledged decommissioning is scheduled to begin after the removal of spent fuel from the spent fuel pool is completed by 2025. Accordingly, many studies related to decommissioning of NPP are in progress, and as part of that, a fleet of container is currently being developed for packaging, transportation, and disposal of various types, shapes, and sizes of wastes generated during decommissioning of the NPP. Since radiation exposure occurs in the process of transporting radioactive materials, the exposure dose assessment for workers and the public should be performed. Therefore, in this study, the risk assessment of transporting decommissioning waste from Kori NPP to Gyeongju disposal facility was evaluated based on the containers currently under development.

In this study, transport scenarios were classified into normal and accident conditions. The normal condition (or incident free) is a transportation condition during which no accident or damage is occurred to the container and considered only external exposure by gamma rays. The accident condition assumes that radioactive material leaks out from the container due to damages to the container by accidents, such as collisions of vehicles, during transportation. When releasing radioactive material by accident, internal exposure was considered with five exposure pathways: inhalation, cloud shine, resuspension, ground shine, and ingestion. Considering population density and traffic volume, transportation routes were set up with national highways and highways in 16 sections. The evaluation target consists of the transport personnel, the workers for loading and unloading of the container, the people along the transport link while the shipment (off-link), the personnel in vehicles sharing the transport link (on-link), and the population at shipment stops for a crew change, passenger transfer, crew meals, refueling, storage, inspection, or any other cause. Each of the six types of packaging and transport containers for decommissioning wastes was applied to the evaluation. The types of waste were low-level RV/RVI, metal, concrete, dry active waste, and soil.

In this study, the RADTRAN code was used for transportation risk assessment and the estimated exposure doses were compared to dose limit for checking the regulatory compliances. According to the results, the effective doses are below the regulatory limits. The study results are expected to be used as a basis for explaining the radiation risk on members of the public and workers on the transport of decommissioning waste.

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Keywords: Decommissioning radioactive waste, RADTRAN, Transportation risk assessment, LILW disposal facility, Annual dose limit

An Exhaust Treatment System That Can Improve Filter Life When Treating Radioactive Waste That Generates a Large Amount of Dust

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A large amount of dust is generated when treating radioactive waste generated by contamination by radioactive materials in operating of nuclear power plants, decommissioning of nuclear power plants, and operating of cyclotron facilities. In addition, a large amount of dust in the case of activation generated by operating of nuclear power plants and cyclotron facilities should be considered. In particular, when treating wastes such as radioactive concrete, a large amount of dust is generated, and a device for purifying air containing dust is essential. Devices for removing dust from air containing dust include a gravity dust collector, a centrifugal dust collector, a scrubbing dust collector, a filter dust collector, and an electric dust collector.

The filter dust collector is mainly used in dry type. In the filter dust collector, if the dust layer of the filter becomes thick, it is periodically removed dust or the filter is replaced. However, when treatment of radioactive waste generated in the radiation controlled area, it is difficult to remove the dust from the filter of the dust collector due to the radioactive contamination diffusion problem, so a filter replacement method must be used.

In this study, we tried to find a way to effectively purify the air containing radioactive waste dust and extend the useful life of the filter than the existing dust collector. Briefly describing this system, first, mist-type water is sprayed in air containing radioactive waste dust before passing through the filter. The air containing radioactive waste dust is primarily removed from the air through mist-type water spraying. And, contaminated water containing a large amount of dust separated from the air by mist-type water spraying is heated to extract dust sludge, and at this time, the evaporated air is designed to be reused during mist-type water spraying through condensation and storage processes.

In the future, the results of this study can be used in nuclear power utilization facilities of the power generation and non-power generation fields where a large amount of dust is generated. The exhaust treatment system will extend the useful life of the filter and reduce the number of filter replacements and the cost of purchasing the filter. Thereby, it will be possible to improve work efficiency and contribute to reducing the cost of radioactive waste treatment and disposal.

Keywords: Radioactive waste, Exhaust treatment system, Dust, Filter, Mist

Study on Application Methodology of DQO Processes for Decommissioning Nuclear Power Plant

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Kori Unit 1 is the first commercial nuclear power plant which was permanently shut down in Korea. Unfortunately, Korea has no experience in decommissioning and dismantling (D&D) project for commercial nuclear except for research reactor. Therefore, it is important to gain an invaluable experience for the following D&D project. For successful D&D project, it is essential not only to develop technologies such as radioactivity measurement, cutting, decontamination, and site restoration, but also to develop a reliable verification methodology for nuclear safety regulators.

The ultimate goal of decommissioning is to prove that the residual radioactivity level meets the site release criteria. To achieve this objective, it is necessary to evaluate radioactive contamination and radiation on the site. Characterization survey is a major activity in evaluating the radiological state of the plant before dismantling. Site characterization plan is designed based on the results of the historical site assessment (HSA) and scoping survey. In the United States, the multi-agency radiation survey and site investigation manual (MARSSIM, NUREG-1575) is recommended as the survey guideline on how to demonstrate using data quality management techniques called data quality objective (DQO) processes. As the U.S. has a lot of experience in decommissioning nuclear power plant, the nuclear safety regulation of Korea is also likely to adopt the same or similar methods for characterization survey plan under MARSSIM. Therefore, DQO processes will be used to verify the detailed characteristic plan.

As the first step to developing the regulatory methodology, we focused on the application method of DQO processes for developing a detailed regulation methodology in characterization survey. For this, the theoretical background for data quality management techniques should be identified and analyzed. Also, the case study will be investigated by securing the regulatory guides and documents for reactors undergoing decommissioning and permanently shut down. These results can be used as basic data for verification of the detailed characteristic evaluation plan. This study will be helpful for establishing detailed regulation methodology for characterization survey.

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Keywords: MARSSIM, Data quality objectives, Characterization survey, Decommissioning and dismantling

A Feasibility Study of Clearance Waste Management Plan

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As the decommissioning of Kori Unit 1 begins in 2023, a large amount of concrete radioactive waste will be generated. In the case of countries that have completed decommissioning nuclear power plants, the cost of radioactive waste disposal accounts for a significant portion of the total decommissioning cost. A large amount of radioactive waste is generated in the process of dismantling a nuclear power plant, most of which are classified as Very Low Level Waste (VLLW), and the amount of radioactive waste in concrete is quite large, accounting for more than 70–80% of the total radioactive waste. The primary shielding wall surrounding the nuclear reactor in a nuclear power plant is generally of reinforced concrete structure, so it is expected that a large amount of concrete radioactive waste will be generated. When the decommissioning of nuclear power plants is promoted in the future, it is predicted that about 160,000 tons of concrete waste will be generated per nuclear power plant. Most of these concrete wastes are VLLW, so they can be considered for self-disposal. Since the level of contamination is very low, the allowable concentration is reduced to the waste subject to self-disposal waste through classification and decontamination, then waste disposal costs can be reduced by landfill, incineration, and recycling.

In this study, I propose an efficient concrete radioactive waste disposal strategy for the reduction dismantling costs. The cost of managing low and intermediate level radioactive waste (LILW) in Korea is 15.19 million won per 200 L drum. This cost is reviewed and updated every two years in accordance with the Radioactive Waste Management Act, and the cost tends to increase by more than 10% each time. This is equivalent to the world's highest cost and accounts for a significant portion of the decommissioning cost. Therefore, through the appropriate concrete radioactive waste disposal scenario, LILW is treated as self-disposal waste to realize decommissioning cost reduction. By examining radioactive waste classification, self-disposal status, and recycling strategies in the US, France, and Germany, which are advanced countries in the field of decommissioning nuclear power plants, it is intended to derive a concrete radioactive waste disposal scenario applicable to the decommissioning process of nuclear power plants in Korea. In addition, based on the results of the self-disposal safety evaluation, we intend to suggest an appropriate disposal method for the bio-shield concrete waste that can be disposed of as a target of deregulation.

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Keywords: Decommissioning, Radioactive concrete, Clearance waste, Recycling of concrete

Applicability of Material Balance Model for Decommissioning Waste to Hypothetical Dismantling Scenario of Steam Generator

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In June 2017, Kori Unit 1 have been permanently shutdown and the preparation for decommissioning of nuclear power plants (NPPs) is actively conducted in Korea these days. Bulk amount of decommissioning wastes from NPPs are generated in short terms, therefore management of decommissioning wastes is important and should be conducted differently from during operation. For ensuring safety and reliability of all decommissioning processes, it is necessary to prove that the total inventory of components before dismantling matches the total inventory of residual components and removed wastes. In this study, the applicability of inventory management of decommissioning wastes has been assessed by applying material balance methodology.

By applying the hierarchical classification system, the spatial information of the decommissioning waste has been presented (e.g., building-floor-room) and the status of the components subject to dismantling have been divided into three categories: before dismantling, during processing and completion of removal from the site. Also, by benchmarking various concept used in nuclear material accounting such as “material balance area (MBA)” and “key measurement point (KMP)”, material balance has been applied to decommissioning wastes. The “waste measurement area (WMA)” has been set variably by the user in consideration of variability, such as the elimination of the rooms and floors due to the demolition of the building during decommissioning by using the concept of MBA. In addition, similar to KMP, the “waste measurement point (WMP)” that monitors the in and out of the components and wastes has been set.

In order to examine the applicability of material balance of decommissioning waste to the actual field, the test case for dismantling, processing and removal of steam generator has been proposed, and virtual simulation on the material balance has been conducted for the test case through the governing equations for mass and radioactivity. This study will contribute to ensuring safety and reliability of all decommissioning processes in terms of inventory management of decommissioning waste.

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Keywords: Decommissioning waste, Waste management, Material balance, Waste measurement area, Waste measurement point

Evaluation of Worker Exposure Dose During Dismantling of AGN-201K Research Reactor

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AGN-201K, operated by Kyung Hee University (KHU), is a lower power reactor for education and training. Tens of AGN type reactors were widely supplied by Aerojet General Nucleonics (AGN) in 1960s. It has been operated for nearly 40 years since it had been transferred from the Colorado State University and had started operation in 1982. The maximum operating power is 10 watt which was uprated from the original 0.1 watt in 2007. The actual operating time of the reactor is usually less than 200 hours a year, much shorter than commercial power reactors.

A research on the lifetime management has been carried out, which included the study on the decommissioning of AGN-201K. As a part of the study, we evaluated the exposure doses that workers will potentially receive during the dismantling process. RESRAD-BUILD was used to estimate the exposure doses. RESRAD-BUILD, developed by Argon National Laboratory, was aimed to estimate the personnel exposure dose rates from the contaminated buildings. A building renovation scenario was taken into consideration including external, inhalation, and ingestion pathways in the scenario.

The location of the sources and receptors are assumed to be inside the building. The source-term including the types of isotopes and their activities was taken from the activation analysis which was already performed using the MCNP computer program. The receptors were assumed to be located at 30 cm from the sources of concrete walls and reactor vessel. The geometry parameters were collected and inputted from the AGN-201K design data.

As a result, the maximum exposure doses of workers were estimated to be 1.04×10^{-2} mSv/yr at the beginning and 9.52×10^{-3} mSv/yr at the end of the decommissioning period. These values were compared with and verified to be much less than the annual dose limit for radiation workers. In conclusion, the potential exposure doses on workers participating in the dismantling of AGN-201K will be negligible. However, the actual exposure doses shall be reevaluated at the time of decommissioning by use of the detected radiation levels. Nevertheless, this study will be useful as a reference for the prediction of the exposure doses of dismantling workers, especially for low and zero-power reactors.

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Keywords: AGN-201K, Decommissioning dose, Worker dose, RESRAD

Development of Blockchain-based IoT Platform for the Radioactive Waste Management

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A blockchain-based Internet of Things (IoT) platform (hereinafter referred to as **RadBlock**) is being developed for real-time monitoring of radioactive waste storage containers that are generated during normal operation or dismantling of nuclear power plants.

The main technology of this RadBlock is a smart IoT platform consisting of a position sensor, a radiation sensor, a temperature and humidity sensor, and an impact sensor for transparent transport of radioactive waste containers from production to final processing and real-time monitoring of abnormal conditions of the containers. In particular, the operation of this platform is based on a blockchain algorithm to secure the security and reliability of data processing that is acquired periodically.

In this paper, we would like to present the measurement performance test results including the configuration of RadBlock's smart sensor system under development, the hardware structure and operating system for data communication and block chain encryption processing process, and the sensitivity and linearity of the radiation sensor.

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Keywords: IoT platform, Blockchain, Radioactive waste container, Smart sensor, RadBlock

Electrochemical Investigation on Primary Loop Base Metals in HyBRID Solution

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HyBRID (hydrazine (N_2H_4)-based reductive ion decontamination) process has been developed for chemical decontamination of primary loop system of nuclear power plants. Oxide film containing radioactive nuclides (e.g., ^{60}Co), deposited on the inner wall of the loop system, is dissolved during the chemical decontamination operation and thus radioactivity of the system can be reduced. In the HyBRID process, N_2H_4 and CuSO_4 are introduced to base solution (e.g., H_2SO_4 solution) to enhance the dissolution rate of the oxide film. Once the oxide film is removed, fresh surface of the primary loop base metals is exposed to the HyBRID solution which is typically acidic. Thus, it is important to investigate chemical stability of the base metals in the corrosive solution. In this study, electrochemical corrosion behavior of several base metals (i.e., 304 stainless steel, SA106 Grade B carbon steel, Inconel 600) was investigated by using potentiodynamic polarization technique. It was revealed that addition of N_2H_4 and CuSO_4 to the H_2SO_4 base solution could reduce corrosion current (i.e., corrosion rate) of the base metals. The corrosion current measured in the HyBRID solution was comparable to or lower than that measured in oxalic acid solution, which is a base solution of CORD (chemical oxidation reduction decontamination) process, suggesting that the base metal corrosion rate of the HyBRID process would be in an acceptable range. The result shows that introduction of N_2H_4 and CuSO_4 in the HyBRID solution offers two advantages on the chemical decontamination: One is the accelerated oxide film dissolution and the other is the corrosion inhibition of the base metals.

Acknowledgements

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Keywords: Chemical decontamination, Primary loop, Hydrazine-based reductive ion decontamination (HyBRID), Corrosion, Electrochemistry, Potentiodynamic polarization

Molten Salt Treatment of Clay Minerals for Cs Removal

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Removing Cs from clay minerals is a key technique for remediation of radioactive soils because of strong radioactivity of ^{137}Cs . In the contaminated soils, Cs is easily combined with the clay minerals because the clay minerals have numbers of stable Cs-binding sites. Aqueous solution treatment has been widely investigated to reduce the radioactivity of the contaminated soils. However, Cs removal efficiency of several non-expandable clay minerals, such as illite ($\text{K}_{0.8-0.9}(\text{Al,Mg,Fe})_2(\text{Si,Al})_4\text{O}_{10}(\text{OH})_2$), was relatively low because of their strong structural stabilities against H_2O molecule intercalation. In this study, high-temperature molten salt treatment was suggested to separate Cs from the clay minerals with high efficiency. The illite samples containing Cs were mixed with $\text{NaCl-MgCl}_2\text{-CaCl}_2$ ternary chloride salt and then the mixtures were heated to 400–850°C. More than 99% of Cs was removed from the illite samples by the salt treatment at 850°C. It was revealed that structural evolution from illite to indialite ($\text{Mg}_2\text{Al}_4\text{Si}_5\text{O}_{18}$), mainly induced by Mg incorporation from the salt, led to extrusion of Cs out of the clay samples.

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Keywords: Soil remediation, ^{137}Cs , Clay mineral, Illite, Molten salt, Chloride

Estimation of Ultraviolet Electric Energy per Mass for Designing Chemical Decontamination Liquid Waste Decomposition System (CWDS)

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In the Preliminary study, KEPCO KPS performed pilot scale test to determine electric energy per order (E_{EO}) for developing commercial scale chemical decontamination liquid waste decomposition system (CWDS). However, E_{EO} is not suitable when oxalic acid concentration is high (e.g., $> 100 \text{ mg} \cdot \text{L}^{-1}$), additional experiments was performed to derive UV-AOP (Advanced Oxidation Process) model based on the electric energy per mass (E_{EM}) to apply high concentration oxalic acid decomposition.

In this experiment, we used pilot scale AOP process equipment, which is consisted of low pressure amalgam UV lamps reactor and medium pressure UV lamps reactor. The simulated chemical decontamination waste water was mixed with hydrogen peroxide in mix tank, and circulated through reactor. UV light was irradiated while the waste water was circulating until the oxalic acid decomposition rate reaches 95%. By sampling at the front and back of reactor, oxalic acid concentration was calculated through TOC analysis.

The pilot test results revealed that the most optimal condition is the waste water with 50 ppm Fe when hydrogen peroxide is stoichiometrically injected both low pressure amalgam UV lamps and medium pressure UV lamps. The results of these conditions are evaluated to determine E_{EM} required to scaling up the UV Lamp reactor. E_{EM} for low pressure amalgam UV lamp reactor and medium pressure UV lamp reactor were obtained each.

The E_{EM} equation can be applied to the standard continuously stirred tank reactor (CSTR) model and integrated to derive the equation for the oxalic acid concentration over time. Assuming the Full System Decontamination parameters, the residual oxalic acid concentration is anticipated at UV-AOP process. By simulating the oxalic acid destruction process with the decomposition rate of 95%, the UV lamp capacity of CWDS was confirmed.

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Keywords: Chemical waste decomposition, UV AOP process, Decontamination, Liquid waste decomposition system, NPP decommissioning

Development of Virtual Decommissioning Core Demonstration Facility for Evaluating Design Requirements for Remote Handling System

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The core demonstration facility for nuclear power plant decommissioning is a research facility to study the material/radiochemical properties of radioactive reactor pressure vessel and internal structures of the decommissioned plant. Recently, the general design requirements and radiological safety evaluation technology for this facility are being developed. One of the key design requirements for this research facility is the requirements for handling equipment for remote operations of equipment installed in the shielded spaces. The requirements include the selection of handling equipment according to the process, accessibility and allowable load of the selected remote equipment, and so on.

In this paper, a virtual core demonstration facility with remote handling devices and process equipment has been developed for evaluating the design requirements. And then, the main process procedures scheduled to be performed at the core demonstration facility have been implemented. Using this virtual facility, the key design requirements for remote handling systems and their auxiliary tools were reviewed. In addition, the adjustment of equipment arrangement and modification of process procedures were performed.

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Keywords: Design requirement, Remote handling systems, Virtual facility, Remote process operation

Studies on a New Treatment Method of Residual HMnO_4 Waste From NPP Chemical Decontamination by Precipitation

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As part of the nuclear power plant primary system chemical decontamination, we studied a new waste treatment method^{KNU patent} for the oxidative chromium oxide decontamination waste including the residual oxidant from the oxidative decontamination process by using a hydrazine. The results of experiments were compared with the conventional method using oxalic acid in terms of amount of waste and radioactive waste stability for final radioactive waste disposal requirement.

It was confirmed that this new oxidative decontamination waste treatment technology has the following excellent characteristics within the scope of this experiment:

(i) The reaction rate of residual oxidative decontamination waste with hydrazine was a very fast precipitate formation reaction.

(ii) The resulting precipitate was Mn oxides with high radioactive waste disposal safety.

(iii) The experimental results presented in this study showed that the generation of decontamination waste was greatly reduced and the disposal stability was also greatly improved comparing with the conventional method.

Keywords: NSSS chemical decontamination, Permanganic acid oxidative decontamination, Oxalic acid, Hydrazine

Application of Chlorination for the Volume Reduction of Radioactive Concrete Waste

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Concrete is one of the largest wastes, by volume, generated during the decommissioning of nuclear facilities, which significantly influences the projected costs for the disposal of decommissioning wastes. Concrete consists of aggregates and a cement binder. In radioactive concrete the radioisotopes are mainly associated with the cement component. If the radioactive isotope can be separated from the concrete to below the clearance criteria then the volume of radioactive concrete waste could be reduced effectively.

This study was conducted to confirm the technical approach of chlorination to enhance the volume reduction process for radioactive concrete wastes. The degree of conversion by chlorination of major radioactive isotopes (⁶⁰Co, ⁹⁰Sr, ¹³⁷Cs) contained in the concrete was evaluated using non-radioactive simulated concrete contaminated samples. The removal efficiency of Cs was confirmed to be over 90% when the temperature was above 800°C, and Co was removed over 95% at 1,000°C. That of Sr was relatively low as 70–80% at 700 and 800°C. The experimental results conducted at various conditions revealed that the temperature is a key parameter for efficient removal of Cs and Co.

Therefore, the chlorination technique could be applied to the volume reduction process for radioactive concrete waste; with options for further process optimization.

Keywords: Radioactive concrete waste, Volume reduction, Flow chemistry, Chlorination

Simulation of Radioactive Liquid Waste From Decontamination of Small- and Medium-sized Metal Wastes of NPP Decommissioning

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Small- and medium-sized metal wastes are generated by decommissioning of nuclear power plants. The corrosion products deposited on these wastes, containing radioactive materials, are removed by using several organic decontamination agents. A large quantity of these organic substances is expected to be remained in the decontaminated liquid waste. Thus, to lower the concentration of these organics up to their acceptable level at the disposal site, the project of “Development of Treatment Process of Organic Decontamination Liquid Wastes from Decommissioning of Nuclear Power Plants” is being carried out. A type of simulated organic liquid waste, supplied to this experimental process for lowering organic substances, was proposed.

Based on the experimental data, from the related several literatures, on the corrosion components removed from the decontamination process of NPPs with organic decontamination agents, the estimated concentrations of Fe, Ni, Cr and Mn ions were 527 ppm, 163 ppm, 151 ppm, and 159 ppm, respectively. In addition, the content of organic substances was estimated to be about 1,000 ppm, and it was assumed that representative organic agents such as oxalic acid, EDTA, and surfactants were mixed at their similar ratios. Also, suspended solids were set at 500 ppm by referring to the liquid wastes characteristics generated during the decontamination process of NPPs. Under these assumptions, the composition of organic liquid waste was simulated, and then organic substances and metal ions were analyzed with TOC analyzer and ICP-OES, respectively. As a result of the preliminary experiment, the TOC was 525 ppm (equivalent organic substances of about 1700 ppm), and the metal ions excluding suspended solids were 171 ppm Fe, 152 ppm Ni, 2 ppm Cr, and 146 ppm Mn, and the total of metal ions was 481 ppm. Most of Cr and some of Fe, expected to exist as ions, might be remained in the suspended solids (451 ppm).

It has been confirmed that the quite large quantities of organic substances are remained and a large amount of suspended solids are also formed in the simulated liquid waste. Therefore, through an additional experiment, the remained organic substances in the simulated organic liquid wastes will be adjusted to about 1,000 ppm and an optimal simulated organic decontamination liquid wastes supplied to the decomposition process of these organic substances will be proposed in the near future. This simulated organic waste would be useful to the development of a decomposition process for lowering those organics.

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Keyword: NPP decommissioning, Organic decontamination liquid wastes, Organic acids, Metal ions

The Establishment of New Package Container Inspection Scenario for Decommissioning Waste in the 2nd Acceptance/Inspection Building

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Currently, the Korea Radioactive waste Agency (KORAD) has been operating the 1st-Phase silo disposal facility to dispose of 100,000 drums of mid and low level waste in the Gyeong-ju disposal site since 2015. Additionally, the construction of a 2nd-phase near surface disposal facility to dispose of 125,000 drums of very-low-level waste/low-level waste is under review for licensing.

When dismantling a nuclear power plant, decommissioning radioactive waste, which is generated in various and large quantities during the decommissioning of a nuclear power plant, is expected to be transported to the acceptance & inspection building in the disposal facility, and disposed after total inspection and sample inspection, as in the current acceptance inspection system. However, the existing acceptance/inspection building within the disposal facility can only acceptance and inspect 200 L and 320 L steel drum waste packages generated from power plants and industries. And the second acceptance/inspection building to be constructed is also designed to inspect steel drums. It is expected that it will be difficult to secure a site and cost for the construction of an additional facility for the acceptance/inspection of decommissioning waste. Accordingly, KORAD plans to receive/inspect and interim storage for large waste packages generated during dismantling of nuclear power plants through design change of the second acceptance/inspection facility.

For large waste package acceptance & inspection, it is necessary to construct and operate facilities such as large cranes (35 tons or more) and new nuclide analysis equipment, and operation scenarios considering interim storage. Therefore, in this study, the decommissioning waste acceptance/inspection scenario (handling, inspection, Tools, storage) was established in consideration of the design characteristics of the second acceptance inspection facility and the waste inspection procedure for the decommissioning waste package under development.

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Keywords: Decommissioning radioactive waste, Acceptance/Inspection building, Total inspection, Sample inspection, Large waste Package, Inspection scenario

Study on Ventilation Systems for a Hot Cell

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A hot cell provides a robust structure, radiation shielding, and confinement for safety of workers and the environment. The hot cell structure is composed of about one meter thick concrete walls; radiation shielding windows, doors, feedthroughs, and penetrations; and remote controllers such as master-slave manipulators. In order to analyze the nuclear power plant decommissioning samples, a hot cell is required to have not only radiation shielding capability, but also function of confinement to filter the airborne releasing materials which are generated during various cutting operations. Several types of cutting instruments could be installed such as a circular saw, band saw, wire saw, CNC lathe and milling machines, and electrical discharge machining in the hot cell. A ventilation system of the hot cell maintains a negative pressure inside its boundary and performs a filtration with HEPA filters to trap airborne radioactive materials or contaminants. To minimize a contamination leaking to the outside, the hot cell is operated at a range of negative pressure between several tens and hundreds of Pa depending on the degree of contamination. In order to maintain a negative pressure, the ventilation system operates continuously, and the emission of radioactive materials is restricted by using a HEPA filter and a charcoal filter. In general, the polluted areas are operated in a once-through manner, and the air flow direction is designed to flow from a low-contaminated area of operating region to a high-contaminated area of the hot cell. In addition, the number of ventilation can be adjusted to prevent the accumulation of suspended substances depending on the contamination producing rate. In this study, the requirements of the hot cell ventilation system are reviewed for an analyzing hot cell of nuclear power plant decommissioning samples. Furthermore, previous research on characteristics of the hot cell ventilation is compared and discussed.

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Keywords: Hot cell, Ventilation system, Cutting instruments, HEPA filter

Uranium Removal From the Contaminated Soil by Using Column Extraction

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Column experiments were conducted for uranium removal using a packed column with uranium-contaminated soil (as a natural material form). A glass chromatography column with a 4.8 cm ID and 15 cm length was uniformly packed with the U-contaminated soil (< 1 m, $30.7 \text{ Bq} \cdot \text{g}^{-1}$ as initial U-238 activity) and initially flushed with distilled water (DIW) to saturate and displace the air from the pore space. Elution was carried out with alkaline and acidic solutions, respectively, and sequentially. The introduced solutions were passed through the column in the upward direction, and the flow rate was controlled by a peristaltic pump. The uranium removal efficiencies were found and a comparison was made with the batch soil washing experiment. During this study, a new approach to rapid column extraction was established, and optimal factors were calculated to demonstrate in industrial-scale uranium decontamination of soil with high uranium content.

Acknowledgements

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Keywords: Soil flushing, Column experiment, Uranium extraction, Uranium removal

A Comparative Review of the Decommissioning Plan for the TRIGA Mark in Korea and the IPR-R1 TRIGA in Brazil

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Decommissioning refers to all activities to be excluded from the application of the law by removing the facility and site or removing radioactive contamination after permanent shutdown. In addition, decommissioning is defined as all administrative and technical actions that should be taken at the end of a nuclear installation in order to assure the suitable physical and radiological protection to the workers, general public, and environment. International Atomic Energy Agency (IAEA) is concerning to establish or adopt standards of safety for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes. In this way the IAEA recommends that decommissioning planning should be part of all radioactive installation licensing process.

The Nuclear Technology Development Centre (CDTN/CNEN) has a TRIGA Mark I Research Reactor IPR-R1 in operation for 47 years. The original power was 100 kW and it is being licensed for 250 kW, and it needs the decommissioning plan as part of the licensing requirements. It has been used for research purpose, neutron beam experiments, radioisotope production, activation analysis and reactor operator training(Angra I and II). The decommissioning plan is the document, in which is organized all information about the proposed decommissioning activities for the facility. It allows the regulatory body to make a proper evaluation and to ensure that decommissioning of the facility can be performed in a safe manner.

Based on the preliminary decommissioning plan of Brazil's research reactor(Included an overview, end state/final goal of decommissioning activities, ongoing activities), it was compared and reviewed for the preliminary decommissioning plan of research reactor in both Brazil and Korea.

Keywords: Decommissioning plan, Research reactor, TRIGA mark, Strategies decommissioning, Project management, Radioactive waste management

Simultaneous Removal of Suspended Soil, Strontium and Cesium From Soil Washing Effluent Using Inorganic Flocculant

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The purpose of this study was to remove suspended soil, Sr^{2+} , and Cs^+ that may exist in soil-washing effluents using an inorganic flocculant (IF). The major chemical constituents of IF were SO_3 and CaO , and were confirmed to have thenardite (Na_2SO_4), calcium carbonate (CaCO_3), and tychite ($\text{Na}_6\text{Mg}_2(\text{CO}_3)_4(\text{SO}_4)$) crystals. When $10 \text{ g}\cdot\text{L}^{-1}$ of IF was injected into the turbid solution containing unstably suspended soil particles of $< 45 \mu\text{m}$, the suspended soil could be completely precipitated. The agglomerates of suspended soil formed using IF had a maximum size of more than 1 mm, due to which sedimentation was effective. The precipitation of suspended soil by IF was more favorable under acidic conditions, such as pH 2, 3, and 5. Sr^{2+} was mainly removed from the supernatant by adsorption using IF, and Cs^+ was precipitated by IF while adsorbed on suspended soil and subsequently eliminated from the supernatant. These results provide that IF can be used for the simultaneous removal of suspended soil, Sr^{2+} , and Cs^+ from a radioactive soil suspension.

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Keywords: Soil washing, Flocculation, Turbidity, Adsorption

Decommissioning LLW Management at Connecticut Yankee Nuclear Power Plant

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Connecticut Yankee nuclear power plant was located in Haddam Neck, Connecticut, that was decommissioned by 2007. The plant was a 560 MW PWR with attendant equipment and facilities. The plant was commissioned in 1968 and operated until it shut down in 1996 for economic reasons. Management of LLW during a decommissioning project has a substantial impact on project timing, cost, duration, and available resources, as well as on disposed waste volumes. Therefore, it is necessary to examine how the major radioactive waste was managed in CY nuclear power plant decommissioning project and to review the lessons learned. Reactor Coolant Pumps & Motors, Primary Loop & Surgical Removal of Contaminated System, Reactor Pressure Vessel, RPV internals, etc. as LLW were the large components removed and dispositioned from CY nuclear power plant. Methods for decontamination, cutting, packaging and disposal of these large components will be investigated and discussed. Class A such as debris and soil was sent to the direct burial. Class B & C such as resins and filters were sent to off-site processing. In addition, the status of ISFSI temporary storage of RPV internals which is GTCC LLW contained in 3 concrete silos will be discussed. Finally, we plan to investigate and share efforts for Tank Farm and Primary Auxiliary Building remediation and groundwater restoration.

Keywords: Connecticut yankee, Decommissioning LLW, ISFSI, GTCC

A Study on Magnetite Dissolution Kinetics for NPP Chemical Decontamination

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As part of the nuclear power plant primary system chemical decontamination, we carried out experimentally a kinetic study of magnetite oxides, which accounts for most of the corrosion oxides subject to decontamination in the nuclear power plants by using a magnetite pellet. The following results were obtained by changing the concentration of oxalic acid for decontamination of magnetite oxide by oxalic acid:

(i) The oxalic acid dissolution reaction rate of magnetite increased in proportion to the oxalic acid concentration.

Keywords: NSSS chemical decontamination, Magnetite pellet, Oxalic acid, Reaction rate of magnetite oxide

Aerosol Filtration Performance Test Facility for Modular Design Based Mobile Air Cleaning System

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Radioactive contaminants such as aerosols can be generated by structure cutting, grinding, scabbling, etc. during the decommissioning of NPP. The aerosols can be suspended in the air and inhaled by the field workers or released outside of workplace. At these decommissioning workplace, the modular design-based mobile air cleaning unit (hereafter mobile ACU) can improve the air condition in the workplace and protect workers from radioactive aerosols.

The mobile ACU consists of filter module, adsorber module and control & monitoring module. The filter module includes a series of bag filter as a pre-filter, metal filter as a medium filter, and HEPA filter considering the conditions of decommissioning environment. This paper presents the test facility to evaluate the aerosol removal performance of the filter module of the mobile ACU.

The aerosol filtration performance test facility includes a shielding room, aerosol generation and aerosol measurement. The shielding room which has a dimension of 3,000 mm (W) × 3,000 mm (L) × 3,000 mm (H) simulates the decommissioning workplace. The PALAS RBG2000D for aerosol generation is used to supply aerosols continuously into the shielding room. The inlet of filter module is connected to the shielding room. The DEKATI HR-ELPI+ is used to measure the aerosol size and concentration at the inlet and outlet of the filter module.

The test is performed with conditions of normal and abnormal situations which include filter loading test. The test aerosols used in the test are Fe₃O₄ and SiO₂ in the size range from 0.1 to 1.5 μm considering the chemical composition of piping and concrete, respectively.

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Keywords: Modular, Mobile air cleaning system, Nuclear power plant decommissioning, Radioactive aerosol, Filtration performance test

Removal of Hexavalent Chromium(VI) in Wastewater Using Chitosan-coated Iron Oxide Nanocomposite Membranes

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Hexavalent chromium (Cr^{6+}) is a kind of toxic heavy metal that frequently exists in wastewater generated from various industries such as steelworks manufacturing, leather tanning, and so on. Particularly, Cr^{6+} is generated in the process of chemical decontamination of nuclear power plant because the oxidized layer of the primary system contains radioactive chromium (^{51}Cr , $t_{1/2} = 27.7$ days). In addition, ^{51}Cr can also be abandoned in wastewaters as a result of several nuclear activities such as radioisotopes production and radiochemistry research. To remove Cr^{6+} in aqueous media, various types of composite materials and processes have been reported for last several years. Among them, membrane filtration is one of the easiest and most effective technique for removal of heavy metal ions.

In this research, we demonstrate the chitosan-coated iron oxide nanocomposite membrane ($\text{chi@Fe}_2\text{O}_3\text{-PVDF}$) for efficient desalination of Cr^{6+} . The $\text{chi@Fe}_2\text{O}_3\text{-PVDF}$ was prepared by a simple filtration method under reduced pressure. This hybrid material was characterized by using scanning electron microscopy (SEM), energy-dispersive X-ray analysis (EDX), and infrared spectroscopy (IR). The adsorption experiments were conducted in both batch and continuous-flow experiments. The experimental data showed rapid removal of Cr^{6+} , which can be described by a pseudo-second-order kinetic model and the Langmuir isotherm model. The nanocomposite membrane exhibited high adsorption capacity for Cr^{6+} ($17.825 \text{ mg}\cdot\text{g}^{-1}$ for batch experiment; $14.104 \text{ mg}\cdot\text{g}^{-1}$ for filtration experiment). Moreover, it maintained high removal efficiency in the presence of several competing ions such as chloride, nitrate, sulfate, and phosphate anions. Consequently, it is expected that the $\text{chi@Fe}_2\text{O}_3\text{-PVDF}$ -based desalination will present a promising strategy and can be developed as a practical method for treatment of Cr^{6+} containing wastewater.

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Keywords: Chromium, Adsorption, Filtration, Iron oxide nanoparticles, Chitosan, Radioactive wastes

Technical Analysis of Radiological Limits for the Open-Air Demolition of Buildings and Structures of NPP

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In order to complete the decommissioning of a nuclear power plant, the DCGL (Derived Concentration Guidance Levels) for the remaining buildings must comply with the criteria established by the regulatory body. However, in general, DCGLs are calculated for the application of soils or remaining building after the decommissioning, which cannot be used as a criteria for open-air demolition.

For buildings and structures contaminated with radioactive materials, the free release criteria for open-air demolition is based on the initial dose criterion (usually, a dose criterion of $10 \text{ uSv}\cdot\text{y}^{-1}$ is used), and clearance levels can be established.

Several decommissioning facilities established open-air demolition criteria based upon the offsite dose impact of the activity from airborne effluents. This approach calculates the acceptable levels of fixed and removable contamination based upon re-suspension factors and ground level release and dispersion models. The objectives of the open-air demolition limits are the following.

(1) Ensure ground level airborne radioactivity levels and off-site dose consequences remain ALARA and within regulatory limits, (2) Ensure demolition liquid concentrations remain at levels which can be collected, processed and released using plant water treatment systems and discharge points, (3) Minimize the spread on contamination within the site boundary such that there is not significant effect on groundwater or the scope of soil remediation required for License Termination, (4) Ensure open-air demolition activities can be conducted using conventional demolition techniques with minimal radiological.

In order to achieve these objectives, it is necessary to distinguish objects based on radiological characterization. Each area can be divided into three categories. R1 was designated to buildings where the radiological contamination is indistinguishable from background. R2 is designated to buildings that reside within the Radiologically Controlled Area (RCA) where the radiological contamination is greater than the R1 levels, but less than the limits specified as being suitable for open-air demolition ($R2_{\text{Open-Air}}$) or interior demolition ($R2_{\text{Interior}}$). R3 is designated to buildings, or portions of buildings, within the RCA where the radiological conditions exceed the $R2_{\text{Open-Air}}$ or $R2_{\text{Interior}}$ criteria, as applicable.

During plant operations and decommissioning all of the open-air demolition objectives have been met by maintaining the loose surface contamination levels below $0.16 \text{ Bq}\cdot\text{cm}^{-2}$ beta/gamma and $0.0033 \text{ Bq}\cdot\text{cm}^{-2}$ alpha on items stored or transported in outside areas. Any materials released off-site are surveyed to less than detectable levels as defined by Nuclear Regulatory Commission Guidance. These levels are less than $0.83 \text{ Bq}\cdot\text{cm}^{-2}$ total beta/gamma contamination and less than $0.16 \text{ Bq}\cdot\text{cm}^{-2}$ removable beta/gamma contamination and less than $0.0033 \text{ Bq}\cdot\text{cm}^{-2}$ loose surface alpha.

Removable contamination is easily locked down by applying a fixative prior to demolition. It was determined that there was already sufficient precedent and experience to justify use of these practices as acceptable targets for $R2_{\text{Open-Air}}$ removable contamination limits. To meet the above objectives removable contamination limits for open-air demolition should not exceed $0.16 \text{ Bq}\cdot\text{cm}^{-2}$ and should on average be less than $0.05 \text{ Bq}\cdot\text{cm}^{-2}$. Tests were conducted at Connecticut Yankee to determine this correlation empirically, rather than by calculation.

This analysis was based on the overseas technical support reports established as the limits for open-air demolition, and it is expected that it can be applied to the decommissioning of nuclear power plants in Korea.

Keywords: Decommissioning, Open-air demolition, Radiological limits, Connecticut yankee

A Study on Basic Data Investigation for Conceptual Design of ILW Storage Containers And Facilities by Decommissioning of NPP

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RVIs, such as baffle plate, former, lower core plate of PWR, and pressure tube, calandria tube of PHWR, installed close to the fuel of reactor core are irradiated and activated significantly by neutron. So, these RVIs are classified as intermediate level radioactive waste after the operation of the facility is terminated, generally.

In the United States, radioactive waste at a level similar to that of domestic ILW is defined as GTCC (Greater-Than-Class C) and stored together in a temporary storage facility for spent nuclear fuel at the site. Interim storage facilities in the United States are commonly referred to as Independent Spent Fuel Storage Facilities (ISFSIs). The United States operates a temporary storage facility at the site of a nuclear power plant, and GTCC temporarily stores it with a separate dry storage container similar to a spent nuclear fuel cask.

Although cave disposal facilities are being operated in Korea, ILW generated from the decommissioning of nuclear power plants has a very high specific radioactivity ($\text{Bq}\cdot\text{g}^{-1}$) of specific nuclides, exceeding the radioactivity limit of domestic cave disposal facilities with only a few hundred kg. In addition, domestic cave disposal facilities can only dispose of LLW and VLLW according to the acceptance criteria.

Therefore, it is necessary to design conceptual model for cask (or container) through the analysis for code and standard of ILW transportation and storage container, and status analysis for ILW storage container and facilities. In addition, storage scenario for ILW generated from decommissioning of NPP should be developed to determine on-site storage options. Because the decision of specification for the ILW container has a very close relationship with the cutting plan for reactor vessel internals and decommissioning of NPP.

In this paper, for the conceptual design of ILW storage containers and facilities, the characteristics and temporary storage status of overseas ILW storage containers and temporary storage facilities were investigated.

Keywords: ILW, Cutting plan, ILW storage containers and facilities, Decommissioning, ISFSI

Waste Management of Primary Bioshield Concrete From ZORITA Nuclear Power Plant Decommissioning

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ZORITA, the first Spanish nuclear power plant, was a 150 MW PWR and operated by Union Fenosa in Spain. The plant was commissioned in 1968 and operated until 2006. In 2010, Union Fenosa transferred the site's ownership to Enresa, a Spanish company responsible for decommissioning the nuclear power plants. The main specifications of the primary bioshield are about 3.46 to about 4.3 m in inner diameter and 7.06 m in outer diameter. The total height is about 8.17 m. The primary bioshield was evaluated for radioactivity using the MCNP-ORIGEN code for a total of 27 operation cycles, and was evaluated as LLW, VLLW and clearance. As a result of evaluating the radioactive properties of the primary bioshield, the low level height and depth were about 2.4 m and maximum 1.32 m, respectively. In addition, the height of the very low level was estimated to be about 6.4 m and the maximum was 1.7 m. The primary bioshield was cut into pieces by dividing the low level boundary radius and the very low level boundary radius into 8 equal blocks using a diamond wire saw. The low level was about 74 tons, the very low level was about 226 tons, and the remainder was subject to clearance. The cut blocks were chopped and packed at the top of the cavity, transported through an airlock to a temporary storage in the turbine building, and planned to be sent to the El Cabril disposal repository. As a result of analyzing the sample by actually coring the primary bioshield, it was confirmed the only about 1.2 m from the inner wall was contaminated.

Keywords: Decommissioning, ZORITA, Primary bioshield concrete, El cabril

Decontamination of Radioactive Ions in Contaminated Soil Using Supercritical CO₂ and Ultrasonic-horn

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Nuclear power generation is used as an energy source in countries around the world, and interest in dismantling nuclear power plants is increasing in Korea in 2017 due to the decision to permanently stop the Kori-1 nuclear power plant. Since Korea has no experience in dismantling commercial nuclear power plants, problems such as unexpected pollution may arise in the process. Therefore, considerable technologies are required to conduct soil remediation of the plant site, especially soil decontamination that can reduce a large amount of radioactive waste is essential. However, since the currently used decontamination methods use chemical solvents or acids, secondary waste is generated. Therefore, supercritical CO₂ that can be separated from the extract due to reduced pressure was used, and ultrasonic horns were added to the process to maximize the efficiency of the reaction.

Supercritical CO₂ is a non-polar material and cannot dissolve metal ions, which are polar materials adsorbed on the soil. Therefore, Catechol Amine, which has excellent bonding strength with metals and acts as a co-solvent, was used. The chelating ligand is bonded with a metal cation to form a metal complex, and paired anions of the metal ions follow together to maintain electrical neutrality. This does not dissolve in supercritical CO₂ and hinders metal extraction, Net4pFOSA, a counter anion that acts as a surfactant, was additionally used to prevent it. In addition, secondary distilled water was also used to promote the flexibility of the ligand and metal ion bonds.

In previous studies, it was confirmed that conditions of 200 bar and 40°C were required for dissolution of reagents, and the experiment was conducted by adsorbing Cs, Sr, and Co ions into the soil and classified by size. Since the decontamination efficiency was high only in the case of sea sand, it was assumed that other relatively small soils would have reacted less with supercritical CO₂ due to their high density. So it was additionally designed to allow supercritical CO₂ to escape into the water in the bubbler through the outlet. Each cycle consisted of generating ultrasonic waves for 3 minutes and then stabilizing for 3 minutes.

In this experiment, a decontamination method using supercritical CO₂ was applied to soil similar to the actual dismantling site. If this process is used, it will be very efficient in terms of economy, such as cost reduction, by reducing the generation of secondary waste.

Acknowledgements

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Keywords: Gori NPP, Soil decommission, Supercritical CO₂, Chelating ligand, Ultrasonic-horn

Status Update for Nuclear Power Plant LLW Disposal Facilities in the United States

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In the United States, the disposal of LLW from nuclear power plants is carried out at four repositories, Richland, Clive, Barnwell and Andrews. The Richland disposal facility is located north of Richland of Washington state and has been operated by US Ecology since 1965. The facility is one of only three full service Class A, B and C LLW disposal facilities in the nation and serves commercial and government LLW customers in the Northwest and Rocky Mountain compact regions: AK, HI, ID, MT, OR, UT, WA, WY, CO, NV and NM. The Clive disposal facility is located in the West Desert of Utah and owned by *EnergySolutions*. The facility plays a vital role in the nuclear industry as a safe and compliant option for permanent disposal of radioactive waste. The facility is also permitted to accept mixed waste, which is a combination of both RCRA hazardous and radioactive waste. The Barnwell disposal facility is owned by the state of South Carolina and operated by *EnergySolutions*. The facility has been operated since 1971 and is the host disposal site for the Atlantic Compact which is comprised of SC, NJ, and CT. The facility is licensed to dispose of Class A, B and C LLW, including irradiated hardware and large components, steam generators, resins, and reactor pressure vessels. The Andrews disposal facility in Andrews County, Texas, is owned by Waste Control Specialists (WCS) LLW and is the only commercial facility in the United States licensed to dispose of Class A, B and C LLW.

Keywords: Richland, Clive, Barnwell, Andrews

Study on the Statistical Decision Criteria for Correlation in Scaling Factor Method

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Scaling factor (SF) method is an empirical method to evaluate the radioactivity of difficult-to-measure (DTM) nuclide based on the correlation between the radioactivity of key nuclides and that of DTM nuclides. The SF method has been implemented in many countries as the principal method for evaluating the radionuclide inventory of radioactive wastes since it greatly reduces the complex radiochemical analysis, which is required to determine the radioactivity of DTM nuclides. Nowadays, interest of the SF method is rapidly increasing due to the essentiality of disposing large amounts of radioactive wastes from nuclear decommissioning in the near future. In the SF method, the correlation between key nuclides and DTM nuclides is evaluated by the Person product-moment correlation coefficient, which is the measure of linear correlation between two variables. Based on the Pearson correlation coefficient, the applicability of the SF method is determined. There are only two decision criteria for applicability of the SF method; using the size of the sample Pearson correlation coefficient (typically greater than 0.6) or using the statistical hypothesis test of the significance of the correlation coefficient (confirming that the population Pearson correlation coefficient is significantly different from zero). The former criteria cannot guarantee that the population correlation coefficient is significantly different from zero depending on the sample size even the sample correlation coefficient is greater than 0.6. On the other hand, in the latter case, even if the population coefficient is not large enough but is significantly different from zero, the SF method may be applied. In the present work, we propose the statistical decision criteria for applying the SF method considering the sample size and the size of the population Pearson correlation coefficient to achieve the more reasonable and advanced decision-making.

Keywords: Scaling factor method, Radionuclide inventory, Decision criteria, Pearson correlation coefficient, Statistics

Review on Current Status of Strontium Decontamination Method

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A large amount of radionuclides was released into the environment due to severe accidents like Chernobyl and Fukushima. Among the radioactive materials, radioactive strontium was especially important because it has long half-life (around 30 years, Sr-90), biophilic characteristics, and high water solubility. Radioactive strontium might have caused contamination of soil, plant, fresh water, and seawater. Strontium acts like calcium. The uptake of radioactive strontium can damage human body for a long time due to accumulation in bone, especially for children. Therefore, the demand for radioactive strontium decontamination has increased. In this paper, the characteristics and limitation of chemical and biological decontamination for strontium were reviewed.

The most of the strontium decontamination study has been concentrated on chemical adsorption and ion exchange method. Chemical decontamination using inorganic, organic, and natural adsorbents like mineral ores and clay were extensively studied. Recent development of nanomaterials on environmental remediation also affect the decontamination study of radioactive materials. Zeolite is the most studied inorganic adsorbent for radioactive strontium decontamination. It has high selectivity, high thermal and chemical stability, and high adsorption efficiency in the condition of seawater and coexistence of potassium and sodium ion. In addition, adsorption efficiency was still maintained after several times of reuse. Graphene oxide has high adsorption capacity due to the abundance of oxygenated functional group like -OH and -COOH.

However, chemical adsorption method is expensive and inefficient for the decontamination of large area with low radioactivity. In addition, decontaminated soil cannot be reused and physico-chemical properties and biological diversity of decontaminated area cannot be restored. For these reasons, cost effective and environmental decontamination method was needed. Biological decontamination method using microalgae and aquatic plant can solve this problem. Microalgae and aquatic plant is cost effective, applicable for large area, and environment friendly. On the other hand, the study for biological decontamination for strontium is in the early stage. Only just screening of microalgae and aquatic plant appropriate for strontium decontamination has been conducted. Thus, biological decontamination method still needed to be developed in the aspect of characteristics, efficiency, and massive production for practical application.

Keywords: Strontium, Decontamination, Adsorption, Microalgae

Deriving Multiplier for the Stress PSF of RPVI Decommissioning Activities

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In previous studies, Task Analysis and Performance Shaping Factors (PSFs) for the Reactor Pressure Vessel Internals (RPVI) cutting work were derived. And Use the Fuzzy-AHP method, which is a quantitative evaluation methodology the PSFs and influencing factors weighting were derived. In this study, the process of determining the multiplier for the stress PSF among the derived PSFs was performed.

Early Human Reliability Analysis (HRA) was performed by calculating the human error probability (HEP) based on expert judgement. Each expert performs a sufficient review of the overall task situation in the process of calculating HEP. However, due to the subjectivity of the experts involved, objectivity may be reduced and differences in HEP calculated by each expert may occur. Therefore, more objective probabilities were derived using PSFs and Multiplier to reduce variability in subjective judgments between experts.

Through previous studies, a clear definition of stress PSF and influencing factors that can affect it was derived. Based on this, an additional review was performed on the stress PSF of the existing HRA literature. As a result, in the case of decommissioning work including RPVI cutting, it was confirmed that when performing this work, the worker's experience, work knowledge, workload, work intensity, and work available time, affect the stress. Working time means the time when a worker performs a normal task, which is evaluated by comparing the time required to perform the task with the time available. In addition, since the degree of stress is affected by work proficiency and knowledge, Multiplier was derived in consideration of these factors. However, in the case of decommissioning work, the work is performed in team units, and the work team is composed of workers with various work experiences based on the procedures. In addition, since pre-education / training of the work is performed, it is judged that there is no impact on Multiplier. And the workload, work intensity, and work available time are specified in the work instruction in consideration of all situations before the start of work. However, due to the overall schedule, actual work time may be limited. In such cases, stress will increase as the work must be completed within the given time in the incident or error situation. Therefore, considering this, the stress multiplier was selected as 4 levels (High Negative Effect, Moderate Negative Effect, Nominal Effect, and Moderate Positive Effect).

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Keywords: RPVI cutting work, Stress, Influencing factor, Multiplier

Immobilization of Radioactive Concrete Fine Powder Waste via Cement Hydration

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A numerous amount of radioactive concrete arises from nuclear power plant decommissioning. In order to reduce the concrete waste volume, multi-step concrete treatment process has been developed in Korea Atomic Energy Research Institute. Concrete fine powder which can be generated during the concrete treatment process must be immobilized due to absorbed radioactive Cs and its dispersibility. This study examined immobilization of concrete fine powder using cement hydration reaction for waste disposal. As concrete waste was heat-treated and pulverized, the hydration ability could be recovered. This feature gave high waste loading for final immobilized product. Effect of additive was considered to investigate the immobilization performance. Also, we focused on the cement concentration in the fine powder, since concrete fine powder can have different Si and Ca contents depending on its origin.

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Keywords: Decommissioning, Radioactive concrete waste, Cement hydration, Immobilization

Safety Considerations Through Regulatory Inspection During Decommissioning Process of KRR 1 and 2

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Since the permanent shut down of Korean 1st Nuclear Power Reactor, Kori Unit 1, interest in decommissioning of nuclear power plant is increasing. In order to achieve the ‘safe decommissioning’ of nuclear power plant, the role of licensee and regulators are important.

Accordingly, it would be a way to adapt overseas NPP decommissioning experience related to decontamination or dismantling technologies and regulatory framework, but a process considering their lesson-learned in a way suitable for Korea will be needed.

It was investigated on the experiences about decommissioning of research reactor 1&2 as Korean 1st nuclear reactor, and was reviewed on the changes of decommissioning inspection since 1997. The review focused on what regulators focused on to achieve the goal of safety decommissioning nuclear power plants, such as inspection methods and items of regulators that have changed as decommissioning progress, will serve as a good reference for verifying the safety of nuclear reactor decommissioning, including Kori Unit 1.

Keywords: Korea research reactor, Dismantling, Decommissioning, Inspection, Safety

Evaluation of the Suitability of Additives Required for Soil Decontamination Using Supercritical CO₂ Through Solubility Measurement

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Today, Korea continues the stance of the de-nuclear policy declared in 2017, and several domestic nuclear power plants are about to decommissioning. Decontamination of power plant sites and nearby soil is one of the essential elements in the process of decommissioning. However, conventional decontamination methods use chemical solvents, etc., and the result after decontamination is generated as secondary waste, so the new methods are needed fundamentally to prevent this. Therefore, in this study, a decontamination method using supercritical CO₂ was applied as an eco-friendly solvent to reduce the generation of secondary waste.

CO₂ is nontoxic and highly permeable, so high treatment efficiency can be expected. It has the advantage of being easy to use because it has a relatively low critical condition with a critical pressure of 73.8 bar and temperature of 31.1°C. In addition, fluid, which is a gas at room temperature such as CO₂, can solve the secondary waste generation problem by separating waste and fluid through decompression, and reduce process cost by circulating and reusing separated CO₂. However, since CO₂ is a non-polar material, additional additives are required to remove polar metal ions that are subject to decontamination. So, the concepts of newly synthesized chelating ligands and co-ligands were introduced.

In this study, a solubility measurement experiment was conducted to maximize the efficiency of the decontamination process by finding the solubility and soluble pressure suitable for decontamination for the chelating ligand among the additives. The temperature condition was unified at 40°C, and dissolution was determined by finding a 'cloud point' indicating dissolution by controlling pressure and volume through a syringe pump and a variable volume device.

As a result of the experiment, several points of dissolution were measured at a pressure of 100 bar or less, and if the optimal dissolution pressure, solubility, and dissolution time derived from the result are grafted to the soil decontamination process, it is expected that the process efficiency will be increased in the future.

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Keywords: Soil decontamination, Supercritical CO₂, Site remediation, Decommissioning nuclear power plant, Solubility measurement, Chelating ligand

Preliminary Study of Dry Segmentation Process of PHWR for Wolsong Unit 1

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As the permanently shutdown of Wolsong Nuclear Power Plant Unit 1 was decided in December 2019, the need for technology for the dismantling of pressurized heavy water reactors emerged.

Accordingly, the CRI of KHNP and KEPCO KPS are developing a technology for cutting and dismantling the CANDU type reactor, the same model as the reactor of Wolsong Unit 1.

The CANDU type reactor is a reactor model that has no reported case of dismantling overseas in accordance with insufficient standards for related regulations and verified dismantling methods. Therefore, continuous research and development in this field must be carried out with close review and evaluation.

In this preliminary study, KEPCO KPS performs to determine cutting plan and process for Wolsong Unit 1 reactor. In the expected plan, predeveloped industrial technologies will be applied, use most of the equipment installed in Wolsong Unit 1 and refrain from using additional equipment. Therefore, practical field applicability and radiation safety considered reactor dismantling process is being developed.

The dry Segmentation method is applied to dismantle the Wolsong Unit 1. Based on the current process of Wolsong Unit 1 pressure tube replacement and its experience and accumulated data, the final amount of generated waste is evaluated to establish standards for Wolsong Unit 1 reactor dismantling.

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Keywords: PHWR, PHWR decommissioning, Calandria, Calandria segmentation, Calandria dismantling

Co Removal From Dissolved Concrete Liquors: M4195 IX Resin Screening Study

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Reactor decommissioning is an important step in the lifecycle of a nuclear power plant to ensure the safe removal of radioactive hazards. A major challenge is the removal and decontamination of the reactor bioshield, primarily because of activation products present owing to a lifetime of exposure to a high neutron flux. Removal of a reactor bioshield generates large volumes of radioactive concrete waste which, if disposed of directly and without treatment, results in a large economic challenge and ineffective space utilization within a disposal facility.

Current surface decontamination methods involve physical and chemical removal of the contaminants. Removal of radionuclides present in the bulk concrete is significantly more challenging. A chemical dissolution method is currently under development for the volume reduction of radioactive contaminated concrete wastes. However, the generation of significant volumes of radioactive Dissolved Concrete Liquors (DCL) pose a substantial challenging to the success of the method. Treatment of these DCLs is of importance to ensure a successful volume reduction and safe release of the treated effluents.

In this presented study, Ion Exchange (IX) has been considered for the removal of cobalt from simulant DCL. The application of IX for the removal of contaminants from waste streams offers a number of advantages including: regenerability and reuse; chemical selectivity; smaller secondary waste volumes; established technology etc. A section of IX resins have been considered and evaluated based on their performance under varying conditions, namely, pH and ion strength. Adsorption isotherms have been studied and evaluated against selected error functions (ERRSQ, EABS, HYBRID, ARE, MPSD) to determine the maximum theoretical loading. Dynamic loading of Co onto a fixed bed column has also been performed at nominal flow rates of 2, 8 and 32 BV Hr⁻¹ to evaluate dynamic loading characteristics and ascertain real world performance metrics, namely, dynamic loading capacities, breakthrough volumes and breakthrough capacity. Several breakthrough models were fitted (Modified Dose Response, Adams-Bohart, Thomas, and Yoon-Nelson) with the Modified Dose Response model giving the best fit using the HYBRID error function. The IX resin M4195 was found to be the most promising candidate for Co removal from DCLs.

Keywords: Decommissioning, Effluent treatment, Ion exchange, Cobalt, M4195

Pre-review of National Reports for the Seventh Review Meeting of the Joint Convention Focusing on Decommissioning Safety

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The Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management is the multilateral agreement of 83 countries as contracting parties. The member states submit the National Reports every three years and assess each other through the question and answer session in the Review Meeting. Although the 7th Review Meeting is postponed to June 2022, the National Reports for this meeting of the member states were submitted to the secretariat in 2020 and some of those are publicly accessible. The reports mainly discuss the current policies, practices, and the general efforts to improve safety with plans of the radioactive waste management and the decommissioning of the nuclear facilities.

The Republic of Korea is now facing decommissioning of the nuclear power plants as Kori-1 and Wolsong-1 were permanently shut down. Since the Republic of Korea has no experience of decommissioning large-scale nuclear facilities, we need to follow up on the latest lessons from countries that already have done the decommissioning and their plan. Furthermore, the different classification of radioactive waste and the options of disposal among the countries should be considered when if the lessons from the other countries are being adopted. Thus, through the analysis of each National Report, this study reviews the current status and the plan of the decommissioning strategy for each member state and analyzes commonalities and differences among the member states.

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Keywords: The joint convention on the safety of spent fuel management and on the safety of radioactive waste management, Spent fuel management, Radioactive waste management, Decommissioning

Preliminary Estimation of Thermal Neutron Flux Distribution in Cyclotron Vault Using Monte Carlo Method

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A few cyclotrons are installed in medical facilities in Korea, producing radioisotopes including ^{18}F . This isotope is usually obtained from the $^{18}\text{O}(\text{p}, \text{n})^{18}\text{F}$ reaction with a ^{18}O enriched water target. This reaction produces neutrons and activate the cyclotron components including concrete walls. After the shutdown of cyclotron facilities, for the site release, licensees perform characterization surveys and classify wastes into RWs and CWs. According to previous research, most RWs are composed of cyclotron components including yolk, dee, target foils, and concrete walls.

In this study, we tried to estimate distribution of thermal neutron flux in vault using MCNP 6.1, and compared it to measurement data, using Au foil and Cd shield. For the experimental measurement of thermal neutron flux, we use 99.99% ^{197}Au foil which has good neutron absorption cross sections, 99% Cd cover which has neutron cut off energy of 0.4 MeV. By positioning Bare gold foil and, Cd covered gold foil in parallel, we can measure thermal neutron flux by distinguishing thermal and epi-thermal neutron flux. To estimate distributions of thermal neutron flux in vault, we attached 7 foil sets on to the entire surfaces of concrete wall with the height of the beam port, other 8 sets with the height of the floor, and the 1 set for near the beam port. By obtaining 411 KeV gamma peak of ^{198}Au using p-type HPGe detector. For energy and efficiency calibration, ^{152}Eu disk check source was used. After, obtaining 411 peaks' net counts of each set and using Cd-ratio, we calculated thermal neutron flux ranged up to 0.4 MeV.

Our calculated thermal fluxes using MCNP codes were ranged same orders with measured. Although there were some differences between them, the trend of distributions between them matched well. Throughout this study, we could check some feasibilities of using neutron transport codes to estimate thermal neutron flux distributions. Also, we could find some implications for modeling and evaluating thermal neutron flux when using MCNP codes, Such as, location of neutron calculation points, any shielding materials in cyclotron vaults. We are looking forward to utilizing it to neutron activation assessment when decommissioning of cyclotron facilities.

Keywords: Cyclotron, Decommissioning, Activation, Thermal neutron, Distribution, MCNP

A Study on the Class Classification of the Hot Cell Facility

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Currently, there are no regulations on the classification of non-reactor facilities in Korea. Accordingly, previous studies confirmed that most of the hot cells operated in the United States were below 10 CFR 100.11 requirements, and considering the characteristics of intermediate-level radioactive waste to be handled by core demonstration facilities (hereinafter referred to as 'the facility'), the facility's structures, systems, and components (SSCs) were classified as non-safety classes.

Based on previous studies, this study provided criteria for the classification of SSCs for safety control of the facility. In addition, after reviewing the classification system of nuclear power plants in Korea and determining the safety class for SSCs included in the facility as non-safety class, it was classified into seismic categories, quality classes, quality groups, and electricity classes by referring to the classification criteria of Korean standard nuclear power plants.

In the case of seismic categories, if the function of treating, extracting, packaging, or storing radioactive waste is performed in accordance with "Nuclear Safety Commission Notice No. 2018-6 (2) 1", the requirements required to perform its own function can be granted through seismic categories. The seismic categories are classified into RW-IIa and RW-IIb according to Reg. Guide 1.143, which corresponds to the seismic category II where functional soundness is not required for safety stop support, but structural soundness is required. For related systems and components those that affect the connection function of the structure are classified into seismic category II, and those that do not are classified into seismic category III.

The quality group is set as the quality group D. It applies to non-safety-related systems and components designed in accordance with the specifications and standards listed in KEPIC MGE (ASME B31.1 applies to overseas purchase items) standards and regulatory guidelines 1.26 and Table 1. Quality group D includes a portion of a system that contains or is likely to contain radioactive materials. Therefore, the facility grants quality group D.

In the case of quality class, the facility is designed by applying quality class A because it is set as a seismic category II structure and a hanger or support related to maintaining the structural soundness of the components.

In the case of an electrical class, since non-1E class electric components are not classified as 1E class, the classes of electric components applied to the facility are defined as non-1E class.

Acknowledgements

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Keywords: Hot cell, Class classification, Design, Non-safety class

A Case Study on Overestimation of Radioactivity Concentration in Scaling Factor Method

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Scaling factor (SF) method is an efficient method to determine the individual radioactivity concentrations. Accurate SF is a key to the accuracy of the individual radioactivity concentrations. Overestimation or underestimation of SFs occurs due to various factors associated with the procedures from the initial sampling to the final data analysis. Although both overestimation and underestimation of radioactivity concentration of difficult-to-measure radionuclides are a significant problem in the implementation of SF method, only the underestimation has been under the strict regulation. Some cases of overestimating the SF and consequently the concentration of difficult-to-measure are investigated as examples of misuse of SF methodology. The issues in this study were limited to the overestimation occurred during the statistical analysis with linear models in SF methodology. Instead of geometric mean, the use of arithmetic mean value of SF is a well-known case of the overestimation. The presence of outlier has a significant impact on the regression analysis, and consequently results in the overestimation of the radioactivity concentrations. In addition, the overestimation from the use of minimum detectable activity (MDA) values as true activity values can induce the severe overestimation. Comparative study on the instrumental techniques shows that thermal ionization mass spectrometer (TIMS) is recommended as a preventive action against overestimation since it can reduce the MDAs. In conclusion, we recommend several countermeasures for the overestimation: (1) use the geometric mean rather than arithmetic mean, (2) evaluate the impact of influential points on the models, (3) do not simply remove outliers and leverages, which many crucial for the data analysis and the future update of SFs, (4) use the MDA values as true activity values with special care only in case of insufficient data for SF implementation, and (5) develop more advanced modeling techniques.

Keywords: Scaling factor, Overestimation, Detection limit, Radioactivity concentration, Difficult-to-measure

Immobilization of Radioactive Sulfate-rich Powder Waste by Low Temperature Sintering

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Radioactive sulfate-rich powder waste can be generated during chemical decontamination process and radioactive liquid waste treatment using ion exchange resin. It is necessary to immobilize the sulfate-rich powder waste into stable form for final disposal. In general, hydraulic cement is applied to immobilize the radioactive powder waste. However, hydraulic cement waste form containing the sulfate-rich waste is unstable due to their expansion, cracks, disintegration problems. Geopolymer can be applied to immobilize the sulfate-rich powder, however, it is not good for volume reduction because of their low density. Iron phosphate glass is also a good binder. However, SO₂ gas is generated during sintering, and large amount of secondary wastes can be generated by SO₂ gas treatment. In this study, low temperature sintering test using low-melting glass was tested in order to immobilize the sulfate-rich waste. From the results, it was verified that the sulfate-rich waste was effectively immobilized into physically stable waste form with high density and high compressive strength. From the PCT and ANS 16.1 leaching tests, it was confirmed that the waste form was chemically durable, and the leachability index satisfies the acceptance criteria (≥ 6).

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Keywords: HyBRID, Decontamination, Sintering, Radioactive waste

Review on Domestic Technical Standards for Decommissioning Waste Transport System

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There are many nuclear power plants (NPPs) with more than 30 operating years around the world, and the demands for decommissioning these NPPs are increasing accordingly. In ROK, starting with the Kori unit 1 in 2017, 12 NPPs are planned to be shut down by 2030. Unlike the wastes produced during reactor operations, decommissioning wastes are characterized by the generation of a large amount of waste in short period of time and various source terms. IP-2 type packages, which are designed and currently used to transport operational wastes, are not suitable for transportation of decommissioning waste because their capacities are not large enough.

Thus, new containers should be designed to efficiently contain and transport decommissioning wastes, and appropriate decommissioning waste transport systems and their technical standards should be newly developed. In this paper, domestic technical standards for decommissioning waste transport systems including transport vehicles and various handling equipment are reviewed for the development of new transport systems and their technical standards. The review focused on whether the 35-ton containers can be transported properly based on the current technical standards for decommissioning waste transport systems and what kind of transport systems and its technical standards should be complemented.

In case of transport vehicles, technical standards for vessel and public roads are reviewed. Domestic maritime transport standards are well established with SOLAS and INF code, which are mandatory international standards, and there were no conflicting standards relating to the 35-ton container transportation. However, maximum allowable total weight for trucks on public road in ROK is 40-ton. Although total weight more than 40-ton could be permitted as an exception, it could be difficult to transport a 35-ton container on public roads considering the weight of a transport vehicle. It is necessary to establish new standards specifically applied to vehicles used to transport radioactive waste on public roads.

In case of lifting crane on a ship, all the lifting gears involved in the lifting must be able to withstand the load of up to 1.25 times of the total weight of the cargo and lifting gears according to domestic regulation. However, the weight of the crane on 'HJ CHEONGJEONGNURI', the dedicated ship for the transportation of radioactive waste in ROK, is only 12.5-ton. It is not enough to lift the 35-ton containers. Thus, the crane on the ship should be replaced or harbor crane should be installed with larger lifting capacity.

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Keywords: Transport system, Domestic technical standards, Container for decommissioning waste, Transport vehicles

Revisit of Simplified Dose Calculation Model in RADTRAN Computer Code

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While shipping radioactive waste, the driver would be exposed to radiation from the package. RADTRAN is a computer code to assess radiation dose and risk from transport of radioactive material, in which 1 m dose rate must be provided as a key critical input parameter. Based on this parameter, RADTRAN calculates radiation dose of the driver in a particular situation. However, 1 m dose rate does not include any information about radioactive source terms at all. Therefore, the validity of using 1 m dose rate as a representative value for all compositions of radionuclides in packages is needed to be discussed.

In this study, an incident free transport scenario in which the package contains only one radionuclide in it was assumed. Concentration of each radionuclide in case of constant 1 m dose rate was calculated using Microshield 9 software. The radiation dose to the truck driver was estimated by using the calculated concentration of radionuclide inducing 1 m dose rate and a pre-established table for dose rate to driver versus unit activity concentration. The calculated dose as such was compared to RADTRAN output for the same scenario and discussed.

Keywords: Transport, Package, 1 m dose rate, Radioactive source term, Radiation dose

System Decontamination Research Projects for Nuclear Power Plant Decommissioning

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In 2013, KHNP prepared the technology development roadmap for PWR decommissioning. The main goal of this roadmap is to completely secure NPP decommissioning technology by 2022, by securing PWR decommissioning design and comprehensive engineering technology and establishing a standard decommissioning technology development model by 2017. However, after KHNP Board decided to permanently shut down Kori-1 NPP on June 16, 2015, the technology development roadmap was revised to secure commercial technologies for Kori-1 decommissioning. In the revised roadmap, 17 decommissioning technologies that have not yet been secured among the commercial technologies required for PWR decommissioning have been identified and one of these, the research for securing full system decontamination technology was completed at the end of September 2018. Through this study, the decontamination process, CRI_RWDecom, and the basic design of commercial facility, for the full system decontamination have been completed. The research for the detailed design and manufacturing of the commercial system decontamination facility, CIDE, is ongoing from 2019.

Wolsong-1 NPP, the first PHWR in Korea, was permanently closed on Dec. 24, 2019. KHNP started a research to secure the technology and conceptual design of full system decontamination for Wolsong-1 NPP decommissioning in 2019 and ended this research on September 30, 2021.

Through this research, it was confirmed that CRI_RWDecom and CIDE for PWR system decontamination can be used sufficiently for PHWR.

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Keywords: System decontamination, CRI_RWDecom, CIDE, Kori 1 NPP, Wolsong 1 NPP

Graphite Electrode Arc Melting Tests by Using Concrete Waste

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A pilot-scale arc melting facility was fabricated to demonstrate the performance to treat hazardous radioactive waste. The facility consists of a melting furnace that melts the waste inside the furnace by generating an arc using three graphite electrodes, a molten salt oxidizer for cleaning harmful and acidic off-gases, and a simple gas treatment system including heat exchanger and HEPA filter.

Arc melting tests were conducted by using uncontaminated and crushed concrete waste because it is one of the major non-combustible radioactive wastes generated during NPP dismantling and it could act as a good matrix for solidifying hazardous materials.

A total of three tests were carried out, two of them were batch type waste melting tests and the last one was continuous waste feeding test. In the batch type test, about 15–17 kg of crushed concrete was loaded inside the furnace before arc generation. And in the continuous feeding test, concrete was additionally fed into the furnace during arc generation and concrete melting.

By applying current to three graphite electrodes, plasma arc was generated at the end of each electrode and then a concrete melting pool was formed within 10 minutes. During operation, the height of each electrode was adjusted up and down and operated at a constant power mode through a change in resistance. The solidified material collected from the inside of the furnace cooled completely after operation was a relatively homogeneous glassy form. About 15 kg of crushed concrete could be completely melted by operating it for about 1 hour with 60 kW of power.

Keywords: Radioactive mixed wastes, Arc melting facility, MSO, Concrete

Review on Decontamination Method of C-14

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Various C-14 decontamination methods (^{14}C -DMs) have been researched, but each method is known to have limits in terms of efficiency, cost, and environment. For this study various ^{14}C -DMs are reviewed, and a comparison analysis of advantages and disadvantages (A&D) of the methods is carried. This paper is aims to discuss the results of the analysis and suggests an ^{14}C -DM to remove C-14 generated from NPPs.

C-14 decontamination generally undergoes desorption, to remove C-14 in a form of $^{14}\text{CO}_2$, then followed by adsorption to recover the desorbed C-14. First, comparison analysis of A&D of main desorption methods, such as acid stripping method (AcSM), alkali stripping method (AlSM), gas displacement method (GDM), thermal stripping method (TSM), and microwave pyrolysis method (MPM), is summarized as follow. AcSM has high desorption efficiency, but other nuclides can be desorbed, making recovery of pure C-14 for recycling to be difficult. AlSM has high efficiency but requires highly concentrated alkaline solution, and furthermore surrounding gas needs to be replaced with inert gas to prevent CO_2 reaction with atmospheric gases, making the overall procedure to be more complex. GDM has lower efficiency compared to previous methods and generates secondary wastes. TSM has low efficiency, but selective desorption of C-14 is possible. MPM is a recently developed method which uses microwave to directly heat up molecules in the wastes, so it has very high efficiency and does not generate secondary waste, but as MPM has insufficient evaluation for verification, it is difficult to be used for actual C-14 decontamination.

Second, comparison analysis of A&D of main adsorption methods, such as physical adsorption method (PAM), chemical adsorption method (CAM), and membrane separation method (MSM). PAM is commercially used method but has low efficiency. CAM commercially uses amine and has high adsorption efficiency, but the chemicals used are expensive and hazardous. MSM allows selective diffusion of $^{14}\text{CO}_2$ from the targeted gas through the membrane, so the efficiency depends on the polymer consisting the membrane. However, as $^{14}\text{CO}_2$ diffuses, $^{14}\text{CO}_2$ and polymer chain undergoes plasticization, thus weakening the durability of the membrane.

When selecting a suitable method, as well as efficiency, other factors such as secondary waste generation, complexity, and usage of hazardous material, should be considered. Although only lab scale evaluation is done, developing MPM and MSM are considered to be the most suitable method for C-14 decontamination, based on the reviewed A&D above, and further development for them is recommended.

Keywords: C-14, $^{14}\text{CO}_2$, Decontamination method, Desorption, Adsorption

6분과

방사선환경 및 안전 (Oral)



Optimization of the Manufacturing Process of Neutron Absorber Made Using 3D Print Through Microstructure Analysis and Improvement

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The existing Boral(B_4C -Al) neutron absorber is made according to the AAR manufacturing method. This method makes a neutron absorber by putting B_4C and Al powder into the basket and then heating and roll pressing it, which has limits that the absorber can only be made in the form of a plate and gaps between absorbers occur. Therefore, experiments and devices using neutrons have some limitations in design, and separation between neutron absorbers becomes a factor that can lead to uncertainty in spent nuclear fuel dry storage container, so a new manufacturing method is needed to overcome this. An objective of this study is to develop a method for manufacturing neutron absorber using 3D printer, capable of overcoming these disadvantages, and to optimize its process.

If a neutron absorber can be manufactured using a 3D printer, the absorber can be manufactured in various shapes so that devices using neutrons can freely design in a direction of improving function and efficiency. It's also possible to eliminate gaps between absorbers that can lead to uncertainty. Of course, in order to fully represent these advantages, the neutron absorber made of 3D printers must be completely manufactured without defects. However, as a result of observing the microstructure using SEM/EDX for some specimens, microcracks with a width of 50–100 μm were detected. And below it, it was found that B_4C components were relatively dense and distributed compared to other places. Such cracks and B_4C aggregation also affected the process of measuring the mechanical properties of the neutron absorber, and the above problems need to be solved to manufacture an intact absorber.

Therefore, In this study, the causes and generation mechanisms of microcracks found in neutron absorbers manufactured with 3D printers were analyzed and proceeded in the direction of improving them. Thermal analysis simulations were used to analyze how aggregation of B_4C components affects the internal structure, and it was confirmed that aggregation of B_4C was the cause of microcracks. After that, to solve the agglomeration phenomenon, the manufacturing process was changed to the 'One Bed Pre-mixing' method, and the absorber specimen was prepared again by optimizing.

As a result of the experiment, it's confirmed that B_4C aggregation and microcracks found in the existing absorber were improved. Through this, If the completeness of the neutron absorber stacked using 3D print is steadily improved, it's expected that a new manufacturing method of neutron absorber and a neutron absorber with improved performance can be developed.

Keywords: Neutron absorber, 3D print, Boral(B_4C -Al), Microcrack, Boron carbide

Comparison of Usage Cases in Nuclear Industry of Hand-held Gamma-ray Imagers

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Hand-held gamma-ray imaging devices have been developed for use in the nuclear industry, nuclear dismantling and decommissioning, radiation safety, non-proliferation, counter-terrorism, and environmental radiation monitoring. In particular, it has the ability to determine the location of real-time radionuclides and analyze them, and additionally has the ability to monitor the radiation dose rate of the workspace in real time. In the case of iPIXTM (Mirion Technologies, Inc.) and Polaris-H (H3D Inc.), which are currently commercially sold, they are used by Korea Hydro & Nuclear Power's nuclear decommissioning technology team in relation to the nuclear decommissioning business. In the case of such gamma imaging equipment, the cases used in Korea are extremely limited and there are no cases of where they can be used. Therefore, we would like to compare and analyze what advantages and disadvantages they have according to the method of implementing the image and examine their functions.

In particular, we will focus on case comparison on the scope of applications that can be used in nuclear power plants during operation, the scope and considerations that can utilize gamma-ray imaging equipment when dismantling nuclear power plants. In addition, for Epsilon-G currently developed at Jeju National University, we would like to present performance comparison and use cases with existing commercial products.

Keywords: Hand-held, Gamma-ray imager, Decommissioning, Nuclear power plants, Epsilon-G

Performance Test of an Airborne Gamma-ray Spectrometer Based on a CZT Detector

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An airborne survey system named the MARK-A1 was developed to be mounted on a commercial drone (Inspire 1, SZ DJI Technology Co., Ltd., China) for the purpose of application in contaminated areas with high dose rate levels. The MARK-A1 system consists of a CZT detector, signal processing unit, and positioning and interface units to a PC on the ground. The weight of the system is below 1 kg for loading on a commercial drone. Two CZT detectors with sizes of $10 \times 10 \times 5 \text{ mm}^3$ (CZT500S) and $15 \times 15 \times 7.5 \text{ mm}^3$ (CZT1500), both from ZRF Ritec SIA, Latvia, were used to construct the small MARK-A1 airborne survey system. From previous study, the irradiation testing and field testing were conducted mainly for CZT500S. In this study we have conducted the irradiation test and field test for both of the two detectors. We derived the G-factor and dose rate calibration factor to calculate ambient dose rates from measured energy spectra with an applicable energy range. An airborne survey was then conducted to check the system performance for about 5 min at heights of 1 m and 10 m near the Gapcheon stream in Daejeon. The ambient dose rate was successfully estimated from the calculation of the measured energy spectra and dose conversion factor. Finally, the results were compared with those from in situ gamma-ray spectrometry using a tripod at the same survey site. The difference between the two systems was below 20%, indicating a good performance of the MARK-A1. It is anticipated that the MARK-A1 can be used to quickly scan wide, highly contaminated areas that are difficult for humans to access in the event of a radiological accident.

Keywords: Gamma-Ray spectrometer, CZT detector, Airborne survey, UAV (Unmanned aerial vehicle), Drone, Ambient dose rate

The Effect of Environmental Factors to Evaluate the Radiation Distribution of Concrete Structures by In-situ Measurement

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More than 70% of the radioactive waste generated during the decommissioning of nuclear facilities is concrete waste, which is an important factor in the economical efficiency of nuclear power plant decommissioning project. Therefore, it is very important to accurately classify of radioactive concrete by radioactivity level. Normally, the radioactivity level of radioactive concrete is performed by laboratory analysis through concrete sampling, but it cannot be evaluated by this method in environmental conditions where regulation and application are not possible. Therefore, it is difficult to apply from the viewpoint of predicting the amount of radioactive waste generated during operation monitoring or decommissioning plan stage. In this study, we optimized the algorithm of evaluation for application to the actual site based on the Peak to Compton method whose validity was verified.

In-situ measurement, factors such as the target and adjacent structure affect the formation of the spectrum. Accordingly, in order to apply the in-situ measurement technology developed in the laboratory environment to the actual field, it is important to accurately understand the factor of the measurement environment. Environmental factors to be considered in in-situ measurement of radioactive concrete are adjacent structures and components, effective area and density. The composition do not affect the Q- β correlation because the activation changes depending on the aggregate mixed in the concrete. On the other hand, angle of collimator and effective area, density are the main factors affecting the measurement spectrum. Therefore, the Q- β correlation for effective area and density was evaluated using MCNP.

In the case of effective area, the effect of Q- β correlation was evaluated by changing the effective area of 1,000 cm $\times$ 1,000 cm based on the 30 cm $\times$ 30 cm plate source manufactured in the laboratory. In the case of density, the Q- β correlation was also evaluated for soil (1.0 g $\cdot$ cm⁻³) which is the main material of a plate source and concrete (2.3 g $\cdot$ cm⁻³), high strength concrete (3.4 g $\cdot$ cm⁻³), super high strength concrete (5.6 g $\cdot$ cm⁻³). As a result of evaluating the effect on the effective area, Q- β correlation was constantly decreasing to a effective area of 500 cm $\times$ 500 cm at the full energy peak (more than 1,000 keV). However, in ¹⁵²Eu with a full energy peak of 778 keV, Q- β correlation was shown in which the measured effective area was constantly decreased up to 400 cm $\times$ 400 cm. If the effective area is large, the Q- β correlation is invalid due to the physical limit of count ratio (Full energy peak and Compton scattering) on the detector. As the density increases, it was confirmed that the Q- β correlation equation also changed because the decrease in full energy peak while the increase in compton continuum by interacting with the medium. Therefore, it is important to confirm the accurate density information about the concrete of the facility to be measured in order to be applied to practical application. Also it is thought that the reliability of the results can be improved by applying the resulting Q- β correlation equation and evaluating it.

Acknowledgements

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Keywords: In-Situ, Radioactive concrete, Peak to compton, Factor, Q- β correlation

Implications of DTM Nuclides in Safe Predisposal Management of Activated Decommissioning Waste

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Kori-1 nuclear power plant is permanently be shut down in 2017 and undergoing the decommission. And it is expected to generate various types of radioactive waste and disposed. In order to the radioactive waste be disposed, there should be needed many kinds of processes to meet the acceptance criteria of disposal facilities. The expected radiological effects to the workers in decommissioning facilities and public who can be affected from the decommissioning should be assessed before the decommissioning.

Especially, because the DTM (Difficult-to-measure) nuclides like H-3, C-14 are difficult to measure the activity without the laboratory analysis, it is difficult to measure the radiological effects of those nuclides on decommissioning sites. Therefore, it is necessary to identify the inventory of DTM nuclides and calculate the dose from decommissioning activities.

During the operation of the nuclear power plants, the materials around a reactor core are irradiated by neutrons and are generated various nuclides and it is the major source of radioactive waste containing DTM nuclides. In this study, the inventory of activation products is carried out with MCNP 6.1 and ORIGEN-ARP. The cross sections of main nuclides were calibrated with neutron flux by the region. The dose calculation is performed by the Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) methodology from International Atomic Energy Agency (IAEA). The SADRWMS methodology is a safety assessment methodology for the predisposal management of radioactive waste. The safety assessment scenarios are containing cutting, transportation, melting, storage. Based on the dose calculation result, a relative risk index for DTM nuclides are made for each decommissioning activity. The risk index relative to the ETD (Easy-to-detect) nuclides of each scenario, enables to quantify the degree of radiological risk of DTM nuclides. This study can be used to support the preliminary decommissioning process for safety assessment of predisposal activities.

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Keywords: Radioactive waste, Decommissioning waste, Activation, DTM nuclides

A Study on the Performance Evaluation of Boron Waste Liquid Separation/Concentration Using Electrochemical Techniques_02

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The boron contained in the waste liquid discharged from nuclear power plants is diluted and discharged into the environment without separate recovery treatment, except for power plants that use boron evaporators. Since the environmental release of boron is directly related to health when ingested in the body, each country is seriously accepting the release of boron into the sea, and as awareness and interest in environmental pollution increase, the regulation on boron is also gradually being strengthened. Boric acid exists in the form of molecules or boron ions (HBO_3^-) depending on the pH. The ion exchange resin is saturated early when used, and the reverse osmosis membrane (RO) has limitations in removing boron in existing nuclear power plant waste treatment facilities due to the limitations of boron separation. Therefore, in the last presentation, EDIXR (Electrochemical Deionization Exchange Resin) technology, which is a boron removal and concentration technology based on ELIX (Electrochemical Ion Exchange) technology, was introduced. In this experiment, it was confirmed that boron can be removed using an electrochemical ion exchange membrane, and the optimum conditions for effectively removing boron by setting various experimental conditions were confirmed. Therefore, in this study, in consideration of the fact that a large amount of boric acid wastewater is emitted from a nuclear power plant, an electrochemical device was scaled up and tested and the removal performance was found to be proportional to the electrode area.

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Keywords: Boric acid, Boron, Electrochemical, Electrochemical deionization exchange resin

Aerosol Removal Efficiency and Life-time Assessment of HEPA Filter From Metal Cutting

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When decommissioning a nuclear power plant, a filtration system is essential to prevent workers from radioactive aerosols. However, the aerosol particles cannot be completely removed by a filter because HEPA filters have the lowest filtration efficiency in a certain size. Therefore, it is still a risk for the internal exposure of the worker. For radiation safety of the worker, the filter efficiency assessment will be required according to the working time.

We assumed that an operator would cut the stainless steel in lengths of 3 cm with a plasma arc torch at a given chamber size. The cutting process repeated the same operation 10 times using the same HEPA filter without exchanging. The HEPA filter (SOLTI, Korea) is a custom-made filter by folding the fabric to form a rectangular shape. The filter was installed between the sampling port to measure the number concentration inlet and outlet passing through the filter. The number concentration of aerosols is measured by using the electrical low-pressure impactor (ELPI+, DEKATI). Aerosol removal efficiency is calculated from the ratio of number concentration at the inlet and outlet filter.

The wind speed, differential pressure, and number concentration of the filter inlet and outlet according to the cutting repetition were measured. Wind speeds were confirmed to be almost unchanged. The differential pressure was confirmed to increase from 45 Pa to 119 Pa according to the cutting repetition. The removal efficiency of HEPA filters was mostly calculated to satisfy 99%. Estimating the lifetime assessment of HEPA filter can help to plan the optimal maintenance period for changing the air filter. This study can contribute not only to advancing safety and improving maintenance planning but also helps to minimize radioactive waste.

Acknowledgements

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Keywords: HEPA filter, Aerosol removal efficiency, Number concentration, Decommissioning, Plasma cutting

Acoustic Emission Data Analysis for Leak Detection in ACPF Mock-up Facility

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Korea Atomic Energy Research Institute (KAERI) has been developing a pyroprocess technology to reduce the waste volume and recycle some elements in a spent nuclear fuel. The process equipment of the pyroprocess uses molten salt, and due to the corrosion problem caused by the molten salt, a hot cell that maintains an inert atmosphere, not an air cell, is required. Therefore, for the efficient development of the pyroprocess, KAERI built an ACPF mock-up facility simulating an inert atmosphere and is currently operating it. The ACPF mock-up facility is that replicates the inert atmospheric control system of the hot cell where the pyroprocessing equipment is operated. The purpose of the ACPF mock-up facility is to verify in advance the remote operation/maintenance of remote handling equipment, cell devices, argon cell control system, electrolytic reduction devices, and other equipment. Because it is installed in a normal laboratory area, not a radiation controlled area, it is convenient to test or operate. The inert atmospheric control system of the ACPF mock-up facility is composed of complex piping, valves, and pumps. In addition, when an experiment using actual spent nuclear fuel is conducted, there is a possibility of contamination of the piping in the system, so system maintenance is very important. However, the complicated piping structure and the large number of valves make this maintenance difficult.

In this study, the leak detection feasibility in the ACPF mock-up facility was verified by using the acoustic emission (AE) technology, which is a technology for detecting leaks in pipes and valves. A detection system was developed to detect the AE signal, and sensors were attached to a total of 8 area to collect the signal, and the optimal parameter value for effectively suppressing the background noise was derived. In order to simulate the leakage of pipes and valves, bypass pipes and valves were newly installed in the existing system of ACPF mock-up facility and defective pipes and valves were installed. The applicability of AE signal analysis technology for ACPF mock-up facility was verified by evaluating the AE signal according to the normal operation and leakage scenarios, and the limitations were analyzed.

Acknowledgements

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Keywords: ACPF, Acoustic emission, Leak detection, Pyroprocess, Inert atmosphere

6분과

방사선환경 및 안전 (Poster)



Development of Pretreatment Process in Tritium Removal Technology and the Behavior of Tritium in the Process

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Tritium (^3H , T) is one of the radionuclides that has the greatest impact on the environment as it is released into the environment along with the operation of nuclear power plants (NPPs). Currently, as ^3H was detected in the groundwater near the Wolsong NPP, where the domestic heavy water reactor is operating, concerns and anxiety about the internal exposure of residents around the power plant are increasing. Because of this, social interest in ^3H contaminated water and ^3H removal technology is rapidly increasing once again. Until now, technologies for removing ^3H from radioactively contaminated water in the system of NPP have been developed and commercialized. However, because these technologies use cryogenic catalytic exchange and/or electrolysis, there are limitations in terms of processing capacity and operating cost. In the case of ^3H polluted water released into the environment, particulate and ionic pollutants that can act as interfering substances must be removed in advance through the pretreatment process in order to be treated with the above commercialized or new technology. In commercial water purification processes with these characteristics, the behavior of ^3H has not been identified. In order to prepare a plan for the treatment of general concentrated water generated by the water purification process, it is necessary to understand the ^3H behavioral characteristics in the process of operating water purification processes. Commercially available conventional water treatment process technologies include Ultrafiltration (UF), Reverse Osmosis (RO), Electro-deionization (EDI), and Capacitive De-ionization (CDI) technologies.

In this study, in order to apply the above water purification technologies to the pretreatment process of ^3H removal technology, we tried to understand the ^3H behavior in each system during the operation of the water purification process. To this end, lab scale equipment of each technology was built. Assuming ^3H environmental release, ^3H simulated contaminated water was prepared using groundwater, river water, and lake water as raw water. Each raw water was contaminated with tritium of $4,000 \text{ Bq}\cdot\text{L}^{-1}$, and it is purified through each equipment and the behavior of ^3H in the system is being investigated. Through this, technologies suitable for the pretreatment process of ^3H removal technology will be selected, and the design to configure the pretreatment process system will be carried out.

Acknowledgements

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Keywords: Tritium, Tritiated water, Liquid radioactive waste, Separation technology, Isotope, Nuclear power plant

A Study on the Analytical Procedure for the Determination of ^{90}Sr in Soil Samples

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In the case of a nuclear accident, environmental pollution may occur due to the release of ^{90}Sr , and the effect of radiation emitted from this nuclide may occur. Since ^{90}Sr has a long half-life of 28.80 years, radioactive contamination by this nuclide continues for a long time. Therefore, in this study, the analysis of ^{90}Sr radioactivity in soil was studied.

In order to analyze the radioactivity of ^{90}Sr , Sr element was separated from the soil sample using Sr-resin, and then beta-rays were measured with a liquid scintillation counter to analyze the radioactivity of ^{90}Sr . In this analysis procedure, the difference according to the amount of Sr-resin required for elemental separation of Sr was investigated. Also, when measuring the radioactivity of ^{90}Sr with a liquid scintillation counter, the results of the Scintillation mode, which measure the beta rays emitted from ^{90}Sr , and the results of the Cerenkov mode, which measure the beta rays of ^{90}Y , the daughter of ^{90}Sr , were compared.

As a result of the Sr extraction experiment according to the amount of Sr-resin, it was found that the gravity method was all separated by 10 ml of elution, and in the case of sequential separation, 20 ml of elution was required. In the case of Scintillation mode, elution should be performed in consideration of using 10 ml of scintillation cocktail in a 20 ml container. As a result of investigating the chemical recovery according to the amount of Sr-resin, it was confirmed that the recovery decreased from 60.30% to 51.96% as the amount of Sr-resin increased from 3 cm to 5 cm.

In the Cerenkov mode, the counting efficiency was determined to be 76.69%, the minimum detectable activity (MDA) of ^{90}Sr was 0.7 Bq L^{-1} , and the recovery was 47.13%. Therefore, it was confirmed that ^{90}Sr measured in the collected soil was smaller than the MDA.

Keywords: ^{90}Sr , Sr resin, Recovery, Scintillation, Cerenkov

Evaluation of Radiological Hazard Parameter of Domestic Building Raw Materials

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Building materials collected from nature all contain natural NORM (Naturally Occurring Radioactive Materials), so residents are continuously exposed to natural radiation. In order to reduce the amount of radiation exposure, related ministries such as Ministry of Land, Infrastructure and Transport, Ministry of Environment, and Nuclear Safety and Security Commission issued guidelines to limit the radiation concentration index to less than 1. The radiation concentration index (EU RP112) is defined as limiting below the reference value through the results of analysis of ^{226}Ra of uranium decay series, ^{232}Th of thorium decay series, and ^{40}K concentrations in gamma emission sources that cause external exposure. It is a method to minimize the amount of radon emission by reducing the concentration of radium. To measure natural radionuclides, uranium decay series ^{226}Ra , thorium decay series ^{232}Th , and ^{40}K were analyzed using a high-purity germanium gamma spectroscopy analyzer (HPGe). The radioactivity level index, radium equivalent radiation, external exposure index, internal exposure index, absorbed dose, and annual effective dose can be calculated from the analyzed radioactivity concentration of the radionuclides (Radiological Hazard Parameter). Studies on the radioactive hazards of such building raw materials are continuing abroad.

This study used building materials used for apartment construction from 2018 to 2020 as samples for analysis. Also, about 10,000 sample samples (Bulk) from 23 countries in Europe were compared with the standards suggested by the UNSCEAR.

Keywords: Radium, Radon, NORM, HPGe

A Methodology of In Situ Gamma-Ray Spectrometry of NORM Residues Based on the MARSAME Manual

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According increasing concerns about NORM residues, IAEA and its member nations have recognized the need for their safety management and guidance in diverse NORM industries. NORM residues with a high radioactivity concentration as forms of scale and sludge are mainly generated during the maintenance of materials and equipment (M&E) in the chemical process of raw materials. The MARSAME manual was then developed as a consensus guidance document to obtain regulatory approval of methodology in the radiological assessment. In this study, a methodology of in situ measurement based on the MARSAME was developed to design and implement a technically defensible survey of M&E. First, the process of building the data quality objective (DQO) and its assessment was reviewed in the data life cycle and the measurement quality objective (MQO) was also reviewed for the reliable measurement technique and assessment of residual radioactivity in M&E. The preliminary survey using a LaBr₃(Ce) scintillation detector was then performed to assess the radioactive range of M&E generated in the domestic mineral process facility of ilmenite ore (iron titanium oxide, FeTiO₃). Gamma-ray peaks induced from natural radionuclides, such as ²¹⁴Bi in uranium decay series and ²²⁸Ac and ²⁰⁸Tl in thorium decay series was definitely recognized during the scan survey. In the process of survey design, the action level (AL) of 0.5 Bq·g⁻¹ within the M&E was selected based on the regulatory body about the definition of NORM residues. The derived limits into surface dose rate from a designated value of the AL was then calculated from the MCNP simulation for simple geometries of M&E. Finally, the feasibility of calculated derived limits for specific M&E was evaluated in diverse geometries of the dimension of M&E, the thickness of sludge inside M&E, and measurement positions using a LaBr₃(Ce) detector.

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Keywords: MARSAME, NORM residues, In situ gamma-ray spectrometry, Uranium, Thorium, Survey design

Dose Evaluation Analysis Before Transferring Liquid Waste Generated in HANARO to the Treatment Facility

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In HANARO, the cause of radioactive liquid waste during normal operation is leakage in the connection part of the system and the piping and the connection part of the equipment containing the radioactive liquid (pump, valve, heat exchanger, etc.).

The liquid waste management system at HANARO collects all radioactive liquid waste, such as equipment and decontamination wastewater, or any wastewater that may contain radioactivity, in the reactor room sump. and the collected waste is transferred to a temporary tank in the isotope building. After that, if a certain amount or more is collected in the temporary storage in the isotope building, it is transferred to the waste measurement facility in the research institute.

The generated liquid waste is transported by pipes or transport vehicles according to the contamination level. As a result, it is not directly emitted from the reactor facility to the environment. Therefore, the radiation monitor for liquid waste is not installed, but the contamination concentration should be measured by collecting and analyzing samples before transferring the waste.

In this paper, there is a recent case of transferring liquid waste from a reactor room sump to a temporary storage tank in an isotope building. At this time, the radiation dose was evaluated before transport and analyzed whether or not the upper limit of detection was satisfied. And effects of liquid waste on the surrounding environment were analyzed.

Keywords: HANARO, Radioactive Liquid Waste, Radioactivity, Sump

Salinity-based Experimental Counting Efficiency Calculation of Main Energy Peaks in the Gamma Energy Spectra of Artificial Radionuclides

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In situ underwater gamma-ray spectrometry is a useful method for monitoring marine radioactivity in real time. However, it is difficult to obtain the counting efficiency of the detector in *in situ* monitoring system experimentally, so the counting efficiency can only be calculated theoretically through simulation mainly without experimental verification. In this study, a method for experimentally calculating the counting efficiency by deriving the radioactivity concentration of ^{40}K , a natural radionuclide, in seawater from the salinity was proposed. The result was compared with the theoretical counting efficiency calculated through MCNP (Monte Carlo N-particle) simulation to verify simulation suitability. As a result, the suitability of the counting efficiency calculation through simulation was verified experimentally with a difference of less than 6%. By calculating the efficiencies for various gamma-ray energies from the same geometry of the verified simulation, the efficiency curve according to gamma-ray energy was obtained. Although this curve itself can be used to calibrate the efficiency of a monitoring system, real-time marine radioactivity monitoring may require a calibration method that is immediately applicable to the gamma energy spectrum obtained by a low-resolution detector, that is, a spectrum where adjacent peaks may overlap. To this end, a method to obtain the ratios between the counting efficiencies for the main peaks of major artificial radionuclides and the counting efficiency for 1,461 keV, the main emission gamma ray energy of ^{40}K , was also proposed. The ratio was called as relative peak efficiency ratio in this study. For theoretical verification, the calculated radionuclide concentration by applying the efficiencies of peaks to the virtual spectra of several major artificial radionuclides and the actual set concentration were compared and the comparison results showed good fitness with a difference of less than 10%. Through the counting efficiency calibration method presented in this study, it is expected that the experimental counting efficiency of ^{40}K and radionuclides of interest can be calculated simply by measuring salinity during *in situ* underwater gamma-ray spectrometry.

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Keywords: *In situ* gamma-ray spectrometry, Experimental counting efficiency, Salinity, ^{40}K , Relative peak efficiency ratio

Bioaccessibility and Absolute Bioavailability of Uranium and Thorium in IAEA-312 by *in Vitro* and *in Vivo* Bioassays

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From human activities such as uranium mining and milling, enrichment, and nuclear power plant (NPP) decommissioning processes, nuclear occupational workers and the public are vulnerable to ^{238}U and ^{232}Th risk. Both ^{238}U and ^{232}Th exposure toward the general population are mainly through oral ingestion. In the case of NPP decommissioning process and severe accident, incidental ingestion of contaminated media could trigger significant risk to animals and humans. Thus, it is a necessary call for toxicity and exposure assessment to quantify the human intake of ^{238}U and/or ^{232}Th in the contaminated media via oral ingestion. In this study, ^{238}U and ^{232}Th bioaccessibility fraction ($^{238}\text{U}/^{232}\text{Th}\text{-}f_{\text{ba}}$) was determined using the *in vitro* Unified BARGE Method (UBM) where IAEA-312 certified reference material (CRM) was used, and later compared to absolute ^{238}U and ^{232}Th bioavailability ($^{238}\text{U}/^{232}\text{Th}\text{-}f_{\text{bv}}$) using *in vivo* Sprague Dawley (SD) model. In gastric (GP) and gastrointestinal (GIP) phases, $^{238}\text{U}\text{-}f_{\text{ba}}$ values were 0.120–0.230% and 0.064–0.125%, and $^{232}\text{Th}\text{-}f_{\text{ba}}$ values were 0.038–0.065% and 0.015–0.024%, respectively. After 360 hours (h), the predicted f_{bv} for ^{238}U and ^{232}Th were both $< 5.00\%$. Both ^{238}U and ^{232}Th were distributed and accumulated in descending order of blood $>$ bone $>$ kidney. Weak *in vitro*–*in vivo* correlations (IVIVC) between UBM and SD-rat assays for blood, bone, and kidney (^{238}U , $r^2 = 0.37\text{--}0.67$; ^{232}Th , $r^2 = 0.20$) was observed. The underestimated f_{ba} results suggested limitations of UBM assay to predict ^{238}U - and $^{232}\text{Th}\text{-}f_{\text{bv}}$ for IAEA-312 soil. The internal dose measurements were much lower than recommended values. Poor IVIVC correlations implied that there is a need for advance exploration of current practices and the *in vitro* methods optimization to support human health risk assessment. These findings can provide the correct exposure estimates for ^{238}U and ^{232}Th in contaminated soil from nuclear facility and radioactive waste management.

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Keywords: Bioaccessibility, Bioavailability, ^{238}U , ^{232}Th

Characterization of Aerosol Released During Plasma Arc Cutting of Cobalt Coated Stainless Steel

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Radioactive aerosol generated from metal waste cutting is a major factor causing internal exposure on a worker. In particular, Co-60 is known as the most hazardous to a worker. And stainless steel shows the highest radioactivity in metal waste. For minimizing the risk of a worker, it is important to understand the characteristic of radioactive aerosol generated during metal waste cutting.

In this study, plasma cutting was operated using stainless steel. The surface of the stainless steel plate was coated with a Co element to a thickness of about 1 μm . A plasma cutter of Hypertherm's POWRMAX125 was used to cut the stainless steel plate. Cutting conditions was set the cutting speed of $10 \text{ mm} \cdot \text{s}^{-1}$, a plasma current of 75A, and total cutting lengths of 30 mm. Metal aerosols generated by cutting a plate were sampled through a sampling tube connected to the chamber. Sampled aerosols were evaluated the number concentration using HR-ELPI+, DEKATI and were quantitatively evaluated by each diameter through ICP-MS analysis. To evaluate the distribution of number concentration for each diameter, the particle distribution variables were calculated using the particle distribution. In the case of CMAD (count median aerodynamic diameter), the diameter equivalent to 50% is the cumulative distribution of the total number concentration values. GSD (geometric standard deviation of particle size) is the result of $\sqrt{((d_{(84.1\%)})/d_{(50\%)})}$, and MMAD (mass median aerodynamic diameter) is $\text{CMAD} \cdot \exp(3 \cdot \ln \text{GSD}^2)$ calculated. The calculation results were calculated as $\text{CMAD} = 0.122869 \mu\text{m}$, $\text{GSD} = 1.366$, and $\text{MMAD} = 0.16441 \mu\text{m}$. In addition, as a result of ICP-MS, the highest concentration of Co element was the range where D50% was $0.155 \mu\text{m}$ in the Impactor collection section, and the mass fraction of Co element was analyzed to be 2.430% in that part.

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Keywords: Stainless steel, Cobalt, Plasma cutting, Aerosol, Number concentration, Mass fraction

Computation of Neutron Attenuation Properties of the Shielding Material in Dry Cask Storage

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At the Korea Institute of Nuclear Nonproliferation and Control, a neutron tomographic technique is currently being developed as a safeguards approach for PWR spent nuclear fuel (SNF) dry storage. In particular, the technique involves a lab-scale demonstration of a novel He-4 scintillation detector array with a model cask and a ^{252}Cf source for measuring both fast and thermal neutrons at the cask surface. The model cask is $1/10^{\text{th}}$ scaled in thickness of an actual ORANO TN-32 dry cask; the canister consists of 24-mm thick inner-layer and 2-mm thick outer-layer made of carbon steels, and 11-mm thick resin placed in between carbon steel layers.

In this work, we compute the neutron attenuation properties of the shielding material in a dry cask via the neutron tomographic technique. We used an array of six Arktis S670e He-4 scintillation detectors to measure different types of neutrons outside the model cask. Besides the $1/10^{\text{th}}$ scaled model cask, we also built a metal frame with an identical dimension to demonstrate the bare-cask environment. The ^{252}Cf was hung at the center of the cask. Then, the He-4 detector array was positioned close to the cask, and emitted neutrons were measured from 0° to 360° at 10° intervals. The identical procedures were carried out with the bare frame for comparison. The back filtered projection was adopted to acquire both the fast and thermal neutron images of the cask and bare-cask cases, respectively.

Despite the 11-mm thick resin layer, the ring shape observed in the thermal neutrons diffuses broader in the bare-cask situation. Also, the number of average neutrons at the cask peripheral has been nearly doubled with the model cask. In contrast, no significant difference in the neutron distribution has been observed for the fast neutron images other than decreased neutron counts. Thus, the influence of shielding material constituting the dry cask should be considered when performing non-destructive inspection of dry cask storage.

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Keywords: Neutron shielding, Dry cask, Fast neutron, Thermal neutron, He-4 detector

Establish of Transverse Tensile Test System for Irradiated Pressure Tube in Hot Cell

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IMEF (Irradiated Material Examination Facility) in KAERI (Korea Atomic Energy Institute) is performing an integrity evaluation of CANDU (CANada Deuterium Uranium) reactor pressure tube. Transverse tensile strength tests were performed to produce data for evaluating the integrity of pressure tube in hot cell.

In the CANDU reactor, the pressure tube is in the calandria tube and coolant called heavy water (D_2O) is following to transfer the heat generated from nuclear fuel. During the initial design life time, the pressure tube is operated under high temperature coolants, high pressure and neutron irradiation. Therefore, the pressure tube becomes brittle due to hazard condition and mechanical integrity is damaged. In order to predict the damage to the pressure tube and to understand the deterioration behavior due to neutron irradiation, transverse tensile test was performed on the tensile specimens manufactured from pressure tube.

Through this study equipment and procedure for tensile test of the pressure tube were established successfully. In order to verify the developed hot cell test equipment and procedure, we performed the transverse tensile test of irradiated pressure tube. In addition extensometer system established in hot cell can be applied to mechanical test of other irradiated materials to produce more reliable data.

Keywords: Transverse tensile test, Irradiated pressure tube, CANDU, Hot cell

Activation Potential Evaluation in Neutron Irradiation Generated by 40 MeV Protons

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The Korea Institute of Radiological & Medical Sciences provides alpha, proton and neutron irradiation services using cyclotron that can accelerate particles up to 50 MeV. The neutrons generated in this service can activate the irradiated subjects. Therefore, in this study, the activation potential of the irradiated subjects by the neutron irradiation service was evaluated with a 10.5 mm thick beryllium target, 40 MeV protons and 20 μ A beam current.

The neutron energy spectrum and the flux in front of the target 1 m were calculated using MCNP 6.2 and the calculated average energy and the flux of the neutron were 13.8 MeV and $1.87 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$ respectively. Under these conditions, iron, copper and tungsten were selected as representative targets and activation was evaluated using FISPACT II. Irradiation time is divided into 15 s, 1 m, 5 m, assuming a routine irradiation, and 1 h, 8 h, assuming the heavy irradiation. Residual specific activity was evaluated immediately after irradiation, and after 10 m, 1 h, 1 d, 7 d, 30 d, 60 d, 90 d, 120 d, 180 d, 1 y, 3 y, 6 y and 10 y.

When iron was irradiated for 15 s, 1 m and 5 m, the total specific activity immediately after irradiation were 3.43×10^2 , 1.04×10^3 and $3.33 \times 10^3 \text{ Bq} \cdot \text{g}^{-1}$, and the fractions by the allowable concentration for self-disposal were 2.98×10^3 , 8.57×10^3 and 2.50×10^4 . In order for the fraction to be less than 1, cooling times of 7 d, 120 d and 3 y were required, respectively. When the irradiation time was 1 h and 8 h, cooling times of 6 y and 10 y were required, respectively.

When copper was irradiated in same conditions, the fractions for self-disposal immediately after irradiation were 1.58×10^4 , 5.71×10^4 and $2.25 \times 10^5 \text{ Bq} \cdot \text{g}^{-1}$ and cooling times of 7 d, 30 d and 1 y were required, respectively. In the sample irradiated for 1 h, after 6 y of cooling time, the fraction decreased to less than 1, but in the case of 8 h irradiation, the fraction was 1.52 even after 10 years had elapsed, exceeding the self-disposal standard.

When tungsten was irradiated in same conditions, the fractions for self-disposal immediately after irradiation were 7.46×10^3 , 1.75×10^4 and $4.14 \times 10^4 \text{ Bq} \cdot \text{g}^{-1}$ and cooling times of 30 d, 90 d and 3 y were required, respectively. In the irradiation for 1 h, after 10 y of cooling time, the fraction decreased to less than 1, but with 8 h of irradiation, the fraction was 1.94 even after 10 years.

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Keywords: Activation, MCNP, FISPACT, Inventory code, Cyclotron, Neutron

Shielding Evaluation of the Spent Nuclear Fuel Handling Hot Cell for ^{252}Cf Sources

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^{252}Cf sources are widely used as a reliable and cost-effective neutron source for activation analysis, neutron radiography, reactor start-up sources, calibration standards, and cancer therapy. ^{252}Cf sources should be handled in shielding facilities such as hot cells because ^{252}Cf emits a large amount of neutrons. In this study, the shielding characteristics for ^{252}Cf sources of hot cells of PIEF (Post Irradiation Experiment Facility) at KAERI originally designed for shielding of gamma rays emitted from spent nuclear fuels were evaluated and the maximum allowable radioactivity of ^{252}Cf in the hot cells was estimated.

MCNP6 computer code was used to evaluate shielding performance of the hot cells for ^{252}Cf sources. The hot cell consisting of heavy concrete wall with a thickness of 88 cm and the lead glass window with a content of 54wt% Pb was modeled using the computer code. ^{252}Cf source of 1 GBq was assumed, and both neutrons and gamma rays emitted from ^{252}Cf source were considered. It is assumed that the source was located at 30 cm away from the inner wall of the hot cell, and the dose rate was evaluated at 30 cm away from the wall of the working area. In order to understand the shielding characteristics of the hot cell, the dose rates by gamma rays, neutrons, and secondary photons by (n, γ) reactions were separately calculated.

In the case of the heavy concrete wall, the dose rate was about 0.6 $\mu\text{Sv/hr}$, of which the dose rate by gamma rays was 0.2%; the dose rate by neutrons was 86.2%, and the dose rate by secondary photons was 13.6%. This is expected to be due to the fact that a large amount of iron ore contained in the heavy concrete dose not greatly contribute to neutron shielding. In the case of the lead shielding window, the dose rate was about 5 $\mu\text{Sv/hr}$, of which the dose rate by neutrons was 98.3%. This result means that the shielding performance of the lead glass for neutrons was much worse than that of the heavy concrete.

Based on the results, assuming that the dose rate criteria in the hot cell working area is 10 $\mu\text{Sv/hr}$, the maximum allowable radioactivity of ^{252}Cf was estimated to be about 2 GBq. If higher activity is required, it is expected to be effective to add neutron shielding materials such as borosilicate glass to the existing shielding windows to reduce neutron dose rates.

Keywords: Radiation shielding, ^{252}Cf , Neutron, Hot cell, MCNP, Dose rates

Redevelopment of Nuclear Emergency Notification System

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Nuclear facilities generally utilize the Emergency Alert System during declaring a nuclear emergency. The purpose of the Emergency Alert System is to quickly notify the emergency declaration to related organizations. The emergency declaration of the nuclear facility should be notified to a number of related organizations. Therefore, if the notification is done manually, time delays and errors may occur. Thus, KAERI runs a Nuclear Emergency Notification System to prepare for the emergency.

It is necessary to contact various related organizations when an emergency declaration of the nuclear facility. The Nuclear Safety and Security Commission, the metropolitan local government, the basic local government, and the Korea Institute of Nuclear Safety are organizations that need to report immediately (within 15 minutes). Besides, a contact network should be established to protect the public including fire stations, police stations, military units, and hospitals.

The nuclear emergency notification system is designed to notify emergency declaration through various means to all the related organizations which are needed to be notified in case of a nuclear emergency. This system utilized multiple means to notify including VMS, SMS, Fax, and Office broadcasting. This system uses the registered terminal to connect to the server and perform emergency declaration notifications. The server is connected to the telecommunication service provider and they call the registered contact network.

The existing system had to be upgraded with the retirement support of Adobe Flash. In the past version of the system, the main contents were designed to be immutable, so it was inconvenient to manage the contact and notification contents.

The redevelopment system is designed to modify notification contents using its own TTS (Text to Speech) engine. Thus, it is possible to flexibly manage the notification contents which can be used during an emergency. In addition, the connection management function of the emergency response organization and the contact system was added. This is designed to reflect the change in the call broadcast in case of changes in the emergency response organization and nuclear facilities.

Nuclear Emergency Notification System in KAERI is operated for effective emergency response in the early phase. This system could enable rapid sharing and response of emergency declaration by simultaneously notification to emergency agents and related organizations. Through this, it is possible to efficiently respond in case of a nuclear emergency and to improve the ability to nuclear emergency response.

Keywords: Nuclear emergency, Emergency alert system, Emergency notification system, Emergency declaration, Broadcasting system

7분과

방사화학 (Oral)



Thermodynamic and Chemical Behaviors of U(IV) Species Under Groundwater Conditions

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Deep geological repositories (DGRs) for the spent nuclear fuel (SNF) are considered as the most effective and safest option for sequestering and disposing of nuclear waste. The performance assessment for the ultra long-term and safe management of DGRs requires an integrated chemical information, including the thermodynamic database of potentially releasable radionuclides from SNF, to examine the models of DGR systems. Therefore, it is important to understand chemical behaviors of major radionuclide of SNF, such as uranium and plutonium, and to measure the reliable thermodynamic data of their various reactions in groundwater systems.

Under the reducing (or anoxic) conditions of DGRs, the oxidation state of uranium is to be lowered so that the chemistry of tetravalent uranium (U(IV)) is prevalent. However, while the thermochemical data of U(IV) has been recognized as one of important parameters for geochemical modelling of radionuclides, the studies reporting those data still have been scarce. In fact, facile oxidation of U(IV) in the presence of atmospheric oxygen often complicates the experiments due to its transformation to U(V) or U(VI) species.

Recently, our research has been directed to examine U(IV) chemical reactions in reducing aqueous environments by thoroughly controlling the experimental parameters such as O_2 levels, pH and temperature, etc. For U(IV) hydrolysis reactions, we improved the reliability of the thermodynamic data ($\log\beta^\circ$, $\Delta_r H^\circ$, $\Delta_r S^\circ$ and $\Delta_r G^\circ$) for the first hydrolysis from U^{4+} to $U(OH)^{3+}$ ions. However, in that study it is concluded that the occurrence of $U(OH)_2^{2+}$ and $U(OH)_3^+$ is very unlikely during the hydrolysis beyond $U(OH)^{3+}$. Instead, we revealed that further hydrolysis from $U(OH)^{3+}$ leads to the formation of U(IV) nanoparticles (NPs), which are clusters (20–30 nm in diameter) of crystalline UO_2 primary particles (typically 5–6 nm). Interestingly, we also found that U(IV)-NPs formation involves the generation of reaction intermediates, presumably, oligomeric oxyhydroxy-U(IV) species, of which formation is kinetically susceptible to the temperature change.

The low solubility of An(IV) species in a wide pH range is well-known and their colloid formation characteristics have been extensively studied in both biotic and abiotic aqueous systems. The U(IV)-NPs prepared in our works further allow us to investigate their colloidal properties and reactivities in the presence of organic molecules. These include organophosphoesters and catechol derivatives, which are selected as model compounds of natural organic materials. Our studies report that the surface of U(IV)-NPs exhibits various reactivities, such as pH-dependent surface acidity originating from the surface hydroxyl groups of U(IV)-NPs, catalytic activity towards organophosphoester decomposition, and the formation of surface complexes with inorganic and organic ions or molecules. The colloidal stability of aqueous solutions containing U(IV)-NPs is found to be highly dependent on the concentration of these organic molecules as well as the solution pH. One of important results from our studies is that the high-pH colloidal stability can be greatly enhanced by surface complexation of organic materials such as catechol derivatives. In the presentation, various spectroscopic evidences supporting these conclusions will be provided and discussed in detail.

Keywords: Tetravalent uranium, Hydrolysis, Nanoparticle formation, Colloid stability, Surface complexation, Organic molecules

Isotope-resolved Atomic Beam Laser Spectroscopy for Detection of Neodymium Isotope

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Nuclear forensics is one of the essential activities for nuclear security and nonproliferation, tracking the origin and history of nuclear materials. Particularly, the identification of physical/chemical properties of target materials is a crucial process in nuclear forensics for the characterization of target materials. In the case of uranium detection, the isotopic ratio of neodymium is an important signature to estimate the history and origin. However, detection of the isotopes in forensic targets typically depends on the complex methods such as mass spectrometry due to their high sensitivity and precision, although they cannot be applied for on-site detection. In this framework, to immediately respond to emergency situations in forensic sites, an advanced detection method is necessary to be developed.

One of the promising solutions for this issue is isotopic-resolved atomic beam laser spectroscopy, which is absorption spectroscopy by using highly collimated atomic beam. In our spectroscopic system, the atomic beam was generated and collimated as the absorption source by adopting the micro-cylindrical crucible. The tunable diode laser with center wavelength of 492.45 nm, which is an atomic transition of neodymium, crossed the atomic beam. After that, the transmission of laser was detected by photodiode with periodically tuning the laser wavelength. It not only provides a high resolution for isotope detection, but also rapid and simple detection procedures. The seven isotopes of natural neodymium could be clearly identified in the absorption spectrum. In addition, several signal enhancement methods, including frequency modulation, thermal reduction, and multi optical pass, were adopted and optimized to improve the sensitivity. Consequently, the detection limit of neodymium was obtained in few-ppm levels in ore matrix. The present study demonstrated the feasibility of our spectroscopic system for nuclear forensics application.

Keywords: Nuclear forensics, Isotope detection, Absorption spectroscopy, Signal enhancement

Application of Pulse Potential Technique for Electrodeposition of Zirconium Metal in Molten LiCl-KCl Eutectic Salt

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Zirconium (Zr) is a key metal element that makes up the nuclear fuel cladding tube. Especially, compared to cheap zirconia (ZrO_2), the value of hafnium (Hf)-free Zr is significantly higher due to the expensive separation process cost because of chemical similarity of Hf. Since Hf-free Zr exists as a cladding material in spent nuclear fuel, by applying electrodeposition process with molten salt system, not only a Hf-free pure Zr metal can be recycled from spent nuclear fuel, but also the volume of low and intermediate level waste (LILW) can be significantly reduced. Particularly, in Republic of Korea, a recycling process of Zr can be operated in a low temperature, low corrosive environment with application of molten LiCl-KCl eutectic salt system based on the development experience of spent nuclear fuel pyroprocessing.

However, due to the complex redox properties of Zr in molten LiCl-KCl eutectic salt compared to other elements, it is difficult to keep steady for certain electrodeposition condition of metallic Zr through both galvanostatic or potentiostatic electrolysis. For this reason, metallic Zr electrodeposited in a molten LiCl-KCl eutectic salt easily form dendritic, low-density structure, and also easily detaches from working electrode due to the poor adhesion strength. This makes hard to collect deposit products as intact form for structural analysis.

In this study, the pulse potential deposition technique, which is a periodical application of electrodeposition potential in a short time interval, is utilized for electrodeposition of metallic Zr in molten LiCl-KCl eutectic salt system. As a result, the adhesion strength of metallic Zr on working electrode is significantly increased, enables to collect high-quality electrodeposits with minimized detachment. Thanks to the pulse potential deposition technique, the structural characteristics of metallic Zr electrodeposited on the tungsten (W) working electrode are studied in detail through scanning electron microscope (SEM). In addition, it was possible to study the reaction mechanism of new reduction peak of cyclic voltammogram, identified only under the conditions of Zr working electrode.

Keywords: Electrodeposition, Metallurgy, Molten salt chemistry, Pulse potential technique, Zirconium

Determination of Non-volatile Radionuclides in Low-and Intermediate-Level Waste From KAERI

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In 2018, KAERI performed the re-analysis of 10 drums randomly selected out of 766 drums that had been transferred to the KORAD yet not permanently disposed of. The re-analysis was carried out throughout the entire process from sampling to the evaluation of radioactivity in the drums. The City Council and the Private Environmental Monitoring Organization in Gyeong-ju audited the drum selection process and sampling. KAERI carried out the radiochemical analysis of the radioactive waste to quantify representative 10 radionuclides and gross alpha including ^3H , ^{14}C , ^{60}Co , ^{59}Ni , ^{63}Ni , ^{90}Sr , ^{94}Nb , ^{99}Tc , ^{129}I and ^{137}Cs . In this study, we present the pre-treatment, chemical analysis, and radioactivity measurement of non-volatile radionuclides such as ^{59}Ni , ^{63}Ni , ^{90}Sr , ^{94}Nb , and ^{99}Tc of the spent filter and corrugated pipe waste. The radiochemical analysis was performed according to the procedure developed in KAERI.

Keywords: Non-volatile, Radionuclide, Low-and intermediate-Level, Radioactive waste

Design of Automated Radionuclide Separation System With Open-bed Resin Column Configuration

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The chemical separation and analysis of radionuclides in radioactive wastes is an essential technology for the disposal of the wastes. Several automated systems have been developed and applied to separate radionuclides in environmental samples with a low level of radioactivity. However, many separation processes dealing with higher radioactivity samples are still carried out in a manual manner based on the gravity flow method and by the use of disposable components to reduce cross-contamination between samples, resulting in laborious and time-consuming workflows. To increase operational efficiency and explore a cost-effective radiochemical separation process, there is a need to develop an automated workflow and its related process equipment.

The conventional automated separation systems have been developed with closed-bed resin column configuration in which the liquid dispenser, resin column separator, and fraction collector are connected in series through lots of selection valves and tubes. In these systems, the tubing pump serves two primary roles: a forced column flow generator and a time-controlled liquid dispenser. The tubing pump has the advantage that the liquid passes only inside the tube, so the pump components do not come into contact with radioactive samples or corrosive acid solutions at all. In addition, the flow rate can be easily adjusted by controlling the rotation speed of the motor, so the configuration of the control system is very straightforward. However, in the case of aging, deformation, and elasticity change of the tube, the flow rate of the tubing pump changes even if the rotational speed of the motor is precisely controlled, which may lead to inaccurate column flow.

In this work, an automated system with open-bed resin column configuration has been proposed to improve some aspects of existing systems. The main parts of the system are separately spaced without connections through any tube and valves. This could help to reduce cross-contamination by minimizing device components in contact with radioactive samples and also reduce the amount of secondary waste from the clean-up process. In addition, an in-line flow rate compensation algorithm has been applied to compensate for the flow rate change due to the characteristic change in physical properties of the tubing pump by adjusting it to maintain the preset flow rate. The feasibility of the proposed concept has been confirmed through preliminary experiments using proof-of-concept equipment with a sensor event-driven controller. These research results will be applied to the development of a movable radionuclide analysis system for nuclear power plants decommissioning waste and will be commercialized after additional performance verification and optimization.

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Keywords: In-line flow rate compensation, Radionuclide separation system, Open-bed resin column, Sensor event-driven controller

Investigation of Metal Oxide for Immobilization of Radioactive Waste

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Many metal oxide materials have been studied for immobilization of radioactive waste. Among the types of materials, zirconolite materials are known as derivatives of pyrochlores that can store several types of cations within its channel. Currently, most of the commonly reported zirconolites based on the formula $\text{CaZrTi}_2\text{O}_7$ (ABC_2O_7) have a structure composed of two types of TiO_6 octahedron and one type of TiZrO_5 polyhedron, which is similar to the cubic pyrochlore structure in the [111] plane. Due to its geometrical advantage, the zirconolite framework has attracted focus as a material that immobilizes the fission products, such as actinides and lanthanides, occurring in high-level nuclear reactor wastes. In this work, we present the synthesis, crystal structure determination, and characterization of a mixed-valent metal oxide $\text{KIn}_{0.33}\text{Te}_{2.67}\text{O}_7$, closely related to zirconolite ($\text{CaZrTi}_2\text{O}_7$), a radioactive waste immobilized material, having a 3D framework. The material has been synthesized as both single crystals and pure polycrystalline phase through hydrothermal reactions. Single-crystal X-ray diffraction calculates that $\text{KIn}_{0.33}\text{Te}_{2.67}\text{O}_7$ crystallizing in the orthorhombic space group, *Cmcm* (No. 63), reveals a 3D framework structure with a 1D channel consisting of Te/InO_6 octahedra and TeO_4 polyhedra. An interesting transition reaction from $\text{KIn}_{0.33}\text{Te}_{2.67}\text{O}_7$ to $\text{KIn}(\text{TeO}_3)_2$ under hydrothermal condition at 230°C is discussed.

Keywords: Immobilization, Zirconolite, 3D framework, Hydrothermal reaction

7분과

방사화학 (Poster)



Automated Chemical Pretreatment for Chemical Analysis of Non-volatile Radionuclides in LILWs

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In case of decommissioning for a nuclear power plant, it is necessary to evaluate radionuclide inventories and radiation dose for decommissioning radwastes whether to dispose of them promptly. For this reason, chemical pretreatment processes are essential to extract non-volatile radionuclides from LILWs (Low & Intermediate Level radioactive wastes). In the pretreatment process, there are incineration, hybrid leaching, and microwave acid decomposition, and etc.

Among these processes, the hybrid leaching has been known as the most efficient method, but it is dangerous because it requires the use of strong acids and works at high temperatures. In addition, there is a disadvantage that a worker can be exposed to radiation for a long time and there may be deviations depending on the worker. Thus, in this study, a fully automated pretreatment system based on the hybrid leaching method for extracting nonvolatile nuclides from radioactive waste was developed to compensate for these disadvantages. The features of the automated pretreatment system are as follows. First, it can be applied with combinations of different solutions for different samples such as soil, cement and solid waste. Second, disposable parts were used to prevent cross-contamination. In particular, the analytical cost was reduced using a reactor made of pp material.

To evaluate recovery rates and reproducibilities of the automated system, SRM 1646A (NIST) was chosen to be the sample since it contains over five nuclides (Ce, Fe, Sr, Ni, Co) whose radioactivity concentration needs to be identified. After the automated pretreatment system was applied, the samples were analyzed with an inductively coupled plasma atomic emission spectrometer (ICP-OES, GENESIS, SPECTRO). As a result of the analysis, the recovery rates were from 88% to 99% according to the radionuclides and the RSDs were approximately 5%.

It was found that there was no significant difference by recovery rate and reproducibility data compared to the existing manual method. Furthermore, it was confirmed that the actual working time of the operator was reduced by 35% compared to the existing method. Based on these results, it is expected that the use of an automated system for chemical pretreatment of radioactive wastes will enable efficient operation by reducing working hours while securing safety of workers.

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Keywords: Chemical pretreatment, Automation, Radioactive wastes, LILWs, Leaching

Estimation of Performance of Automated Separation System for Mixed ^3H and ^{14}C in LILWs

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In general, ^3H and ^{14}C in LILWs (Low & Intermediate Level Radioactive wastes) should be separated and measured their radioactivities. In particular, in case of dismantling for a nuclear power plant, it is necessary to quickly separate radionuclides and accurately measure the radioactivity for the judgment of waste disposal. Chemical separation of volatile nuclides using oxidizing agents is mostly done manually, using very strong acids and at high temperatures. Therefore, the workers can be exposed the danger of acids and high temperature, and human errors can be often occurred depending on the skill and fatigue level of the worker. In addition, a lot of time is required for cleaning, rinsing and drying of glassware, and residual contamination is also a concern. In this study, fully automated system was developed and evaluated for the separation of ^3H and ^{14}C nuclides to compensate for these disadvantages, and compared with the manual method.

$\text{K}_2\text{S}_2\text{O}_8$ and AgNO_3 for the oxidant with the standard materials H_2SO_4 with nitrogen gas were introduced in a reactor. And then, $^{14}\text{CO}_2$ for ^{14}C was trapped to the absorber (Carbo-sorb E, Perkin-Elmer), and tritiated water (HTO , $^3\text{H}_2\text{O}$) was distilled to the collector for ^3H . The ^{14}C (#6014, Eckert & Ziegler) solution and ^3H (#6003, Eckert & Ziegler) solution as standard, and Permafluor E^+ (Perkin-Elmer) for ^{14}C and Ultima-Gold LLT (Perkin-Elmer) for ^3H as cocktail were introduced. The beta-ray of extracted ^{14}C and ^3H were counted by the liquid scintillation counter.

The recovery rate of ^{14}C and ^3H by automated system was estimated as 97% and 98% (5 times average with 3.66% and 2.88% as root mean square) compare to the certification of standard solution, respectively. And the recovery rate by manual method for separation was shown as 98% for ^{14}C and 97% for ^3H with 3.19% and 2.88% as root mean square.

From the result, it was found that there was no significant difference between the automatic method and the manual method. And it was also known that the actual working time for such as heating on/off and pre- and post-processing was dramatically reduced, regardless of the skill level and fatigue level of the workers. Therefore, the introduction of the automation system is expected to improve the safety of workers in exposure, reduction of human error, and improvement of data reliability.

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Keywords: ^3H , ^{14}C , Radioactive waste, Wet oxidation-acid, LSC

Modeling Deposition and Erosion of CRUD on Cladding Surface in PWR

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CRUD (Chalk River Unidentified Deposit or Corrosion Related Unidentified Deposit) is a thin oxide layer, growing up to about 100 μm , on the surfaces of fuel cladding in PWR (Pressurized Water Reactor). This layer has been researched as a cause of AOA (Axial Offset Anomaly) through boron hideout and as a candidate for undesirable thermal resistance during normal and accident conditions. High pressure and high temperature environments for CRUD deposition hinder obtaining reliable experimental observations. For this reason, there is no well-defined model for the deposition and erosion mechanism for CRUD layers.

In this study, we have developed a new CRUD deposition and erosion approach to simulate deposited CRUD layers on cladding surfaces under PWR operating conditions. Wick-boiling structure, a spatial characteristic of porous CRUD under PWR conditions, leads to evaporative-driven heat transfer and capillary flow for convective heat transfer. The thermal conductivity model simulates the effective thermal conductivity based on the CRUD composition and flow properties within the deposits. During the deposition process, the particulate flux determined by sub-cooled nucleate boiling accelerates the deposition amount when the CRUD grows. Diffusional flux and turbulent flux were assumed as negligible, due to the dominant portion of boiling-enhanced flux toward the fuel surface. Perfect sticking was assumed according to the nature of fine particle size on the clean surface because almost CRUD particles show a size near 0.2 μm . For erosion mechanism, the erosion coefficient from Beal's correlation, calculated by drag force applied on the surface particles, links the turbulent kinetic energy and erosion rate. Drag force was calculated by using shear stress, which is linearly proportional to the turbulent kinetic energy near the surface. Through the Newton-Rapson method, CRUD deposition coupled to heat transfer and capillary flow simulates the full-cycle CRUD growth for 1 fuel cladding along an axial direction.

In these results, almost 40–50 μm of CRUD was deposited under conditions of Seabrook Unit 1 Cycle 5. Sensitivity study was conducted for the geometrical properties and composition of the deposits. This model is fully based on the theoretical base without empirical constant used to match the deposition pattern and degree of erosion for plant data was used.

Keywords: Sub-cooled nucleate boiling, Deposition, Turbulent kinetic energy, Erosion, CRUD

Performance Testing Results on Tritium Determination in the Presence of C-14

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GHB-15, which consists of graphite heating blocks, distillation beaker, and watch glasses, is a commercially available distillation kit for tritium determination in aqueous solution. Previously, it was confirmed that there was no cross-contamination and tritium was quantitatively recovered. Due to a characteristic of liquid scintillation spectrum, other volatile radioisotope has an effect on a quantitative analysis for tritium. So, it is worth to check whether it is able to purify tritium in aqueous solution. To check the above mentioned, 3 different simulant solutions were prepared as follows: simulant solution A is composed of tritium and C-14 in concentration similar to tritium, simulant solution B is composed of tritium and 5 times higher concentration of C-14 than that of tritium, simulant solution C is composed of tritium and 10 times higher concentration of C-14 than that of tritium. Afterward, each simulant solution was distilled using GHB-15 and the tritium concentration in distillate was determined by liquid scintillation counter.

According to performance testing results, GHB-15 was good at discrimination between tritium and C-14 in comparison with untreated simulant solution and so it was confirmed that the tritium in distillate was quantitatively recovered.

Keywords: Tritium, C-14, Distillation, Performance testing results

Efficiency Calibration of C-14 for LSC – It's Problems and Resolutions

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Carbon-14 is one of fourteen radionuclides which must be identified from the radioactive wastes in order to get stored in a repository. C-14 also needs to be routinely monitored in air samples from NPPs. Liquid Scintillation Counters (LSC) are the most widely used instruments for C-14 measurements. Prior to the routine measurements, the LSC should be properly calibrated for C-14 using a set of traceable calibration standard.

It is common that the users use a set of commercially available calibration standard but those sets are not exactly same in geometry and chemical compositions etc. as those of actual samples prepared at laboratories for counting. If users want to use the set of commercially purchased standard for calibration, it is advised that such calibration details should be validated prior to the routine use.

Radsol Co., Ltd. have been using the in-house standard sets for C-14 efficiency calibration and have recently noticed a problem. The measured C-14 values of calibration sets were not stable over time which could lead to a significant failure of C-14 data. Two different chemical forms of C-14 standard, carbonate and toluene, were used to prepare the calibration standard sets in glass vials using CarbonTrap trapping agent & CarbonCount cocktail supplied by Meridian, UK. The prepared calibration standard sets were counted in a HIDEX 300SL liquid scintillation counter. The problem was investigated over a period of time and the findings are presented here.

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Keywords: C-14, Efficiency calibration, Liquid scintillation counter, Carbonate, Toluene

Development of Simultaneous Separation Method for Alpha Radionuclides

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Nuclear forensics is a major analysis technique for nuclear materials using the decay characteristics of radioactive materials. Abroad, the International Atomic Energy Agency (IAEA) is using the term, Nuclear Forensics, to develop analysis technology and to interpret analysis results. In Korea, radionuclide analysis is mainly performed by government agencies and government-funded research institutes. Other commercial laboratories are not very well equipped and have limited capabilities.

It is necessary to develop an alpha radionuclide analysis method for an unknown sample to find out whether radioactivity is present and the origin of such radioactivity. In order to separate U, Th, Pu and Am, which are the major alpha radionuclides, standard solutions of each radionuclides were added to ultrapure water to conduct an experiment. Anion and UTEVA resins were used for the method development. Th and Pu were separated from the anion resin and U was separated from the UTEVA resin. Am was separated using the final filtrate. The amount of iron carrier for co-precipitation was optimised for the highest chemical recovery of all tested radionuclides. In addition, the amount of acids for separation was also optimised in order to minimize the acid wastes. Final results showed that the chemical recovery of all tested radionuclides, U, Th, Pu and Am, were higher than 80%. The developed method was finally validated using IAEA-384 standard reference material.

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Keywords: Nuclear forensic, Radionuclides separation, Uranium, Thorium, Plutonium, Americium

High-precision Distance-corrected Laser Monitoring Under High Temperature Molten Salt

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Molten salts are widely considered for future nuclear applications including molten salt reactors and pyroprocessing. The main characters of these systems are high temperature and strong radiation. When considering the two harsh environments, the monitoring system needs to detect nuclides at a long distance for avoiding thermal and radiation damages. In addition, a monitoring technique should have enough resolutions to distinguish many different elements simultaneously. As a viable option, laser-induced breakdown spectroscopy (LIBS) has been considered by several research groups especially in U.S.A and U.K. However, the previous studies showed unacceptable tendency of the calibration curve and relative standard deviations (RSD) from 30–100%. In particular, the measurement of lens-to-sample-distance (LTSD) is one of the challenges to using LIBS monitoring. The consistency of LTSD determines the precision of measured data. Many existing techniques to measure LTSD is not applicable since high-temperature molten salts are transparent liquids. The most advanced method using the electric resistance between two electrodes equipped with a step motor cannot reach acceptable sensitivity for LIBS measurements.

In this work, to overcome such limitations, we developed a LIBS monitoring system equipped with a simple electronic circuit for using an electret condenser sensor. The electret condenser sensor can detect the amplitude of shock waves that arise from laser-induced plasma. While it does not measure exact LTSD, it can correlate shock wave amplitude and LIBS intensities. All instruments are synchronized by the delay generator so that the shock wave is detected by the sensor with LIBS detection simultaneously. The amplitude of the shock wave and the LIBS intensity were recorded with different LTSD. Furthermore, the calibration curve for Sr by LIBS techniques was recorded with three different LTSD for checking the effectiveness of the consistent LTSD. The experiments were controlled with 600°C for LiCl-KCl using a glassy carbon crucible. In these experiments, we successfully got the acceptable correlation and the large difference of the monitoring sensitivity results with different LTSD. The RSD has been improved to 3–7%. The high linearity of the calibration curve was showed at the best focal point that was extracted by the electret condenser sensor. However, an unacceptable tendency of the calibration curve was shown at the improper focal point.

Acknowledgments

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Keywords: Laser-induced breakdown spectroscopy, Pyroprocessing, Molten salt reactor, Safeguards, Process monitoring

Chemical Thermodynamics Study of Aqueous Am(III) Complexations With Isosaccharinic Acids

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Am(III) is one of the most concerned long-term radiotoxic radionuclides and its chemical thermodynamic properties should be understood in natural groundwater environment for the safety assessment of the repository systems for radioactive wastes. Isosaccharinic acid (ISA) is a major cellulose degradation product and is an abundant natural organic ligand. ISA contains a carboxyl group and four hydroxyl groups that cooperatively interact with cations to form strong complexes in nature. OECD NEA chemical thermodynamic database project pointed out the important roles of organic ligands including ISA on transporting radionuclides through geospheres. Accumulating more data on actinide-ISA interactions are in demand due to the lack of available data, in particular on the trivalent actinide and lanthanide ions. In this study, we investigated Am(III) interactions with ISA by using UV-Vis absorption spectroscopy, time-resolved laser fluorescence spectroscopy (TRLFS), and DFT calculations. Characteristic changes of the absorption spectra of Am(III) were observed as a function of ISA concentrations. Quantitative analysis on the absorption spectral changes shows formation of 1:1 and 1:2 Am(III)-ISA complexes. TRLFS results indicate tridentate bindings of each ISA ligand. DFT calculations suggest that 2- and 4-hydroxyl groups as well as the carboxyl group are interacting with Am(III) atom and at least one of the two hydroxyl groups is bound as a deprotonated state. Temperature-dependent equilibrium shifts were examined in the temperature range of 15–75°C. For 1:1 complex formation, the reaction is endothermic driven by entropy increase. On the other hand, 1:2 complex formation is shown to be exothermic. Determined chemical thermodynamic properties of Am(III)-ISA complexes will be discussed in relation with the spectroscopic properties and the proposed structural information.

Keywords: Americium, Isosaccharinic acid, Coordination chemistry, DFT calculations, Spectroscopy

Temperature Effect on the Adsorption Behavior of U(VI) Onto 2-line Ferrihydrite

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2-line ferrihydrite is an abundant amorphous iron oxide with a high specific surface area compared to other iron oxides. This mineral property suggests that it can be used as an adsorbent, and the adsorption of the hexavalent uranium, U(VI), onto 2-line ferrihydrite has been extensively studied. However, most studies have been performed near room temperature, even though higher temperature is expected in deep geological repositories for nuclear waste due to its decay heat and the geothermal gradient.

This study examined the applicability of aqueous U(VI) removal by 2-line ferrihydrite. 2-line ferrihydrite was synthesized by titrating a solution of $\text{Fe}(\text{NO}_3)_3$. Prepared 2-line ferrihydrite was characterized using X-ray diffraction (XRD), surface area analyzer, and scanning electron microscopy (SEM). The adsorption properties of U(VI) onto 2-line ferrihydrite were investigated for the influence of various chemical parameters: pH, contact time, S/L ratio, and temperature. The U(VI) adsorption onto 2-line ferrihydrite was determined by measuring the uranium concentration of the supernatant using inductively coupled plasma optical emission spectrometer (ICP-OES) and mass spectrometer (ICP-MS).

Acknowledgements

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Keywords: Adsorption, Uranium(VI), Ferrihydrite, Effect of temperature

Modeling of Fission Product (Cs, I, Xe, Tc) Release in UO_2 Oxidized to U_3O_8

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In KAERI, for the development of nuclide management technology reducing disposal area required for spent fuel management, optimal thermal treatment condition related to fission product release is being developed; mid-temperature and high-temperature treatment conditions. To avoid the formation of cesium pertechnetate not easily trapped in off-gas treatment system, in the former treatment, technetium, selenium and krypton should be mainly released and cesium and iodine be released in the latter treatment.

In determination of such condition, due to a difficulty of spent-fuel experiment, a strategy employing model prediction was adopted and two-stage diffusion release model coupled to UO_2 oxidation model developed. For model development, the following assumption were made:

- All pellets are approximated to sphere having same ratio of surface to volume.
- Cracking occurs due to the formation of U_3O_8 occurs in solid surface yielding to the increased reactive surface area.
- Porosity of pellet changes as reaction proceeds.
- Oxygen diffusion follows 1D spherical diffusion.
- Sphere is equally divided into symmetrical shell.
- All fission product exist in grain body and gaseous diffusion through an open pore follows the diffusion through a solid spherical fuel grain.

In our previous study, due to a limited fission product release data, the modeling of cesium and krypton release behavior was mainly focused. However, in this study, due to availability of technetium and iodine release data, the developed model is applied to the modeling of technetium and iodine release behavior besides cesium and krypton by regressing model parameters to the experimental data. It was found that the developed fission release model closely describes the experimental behavior of fission product of interest, implying a validity of model prediction.

Acknowledgements

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Keywords: Two stage-diffusion model, Fission product release, Diffusion coefficient, UO_2 oxidation, One-dimensional

High-Temperature Viscosity Measurement of KNO_3 Melt Using Rotational Viscometer

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Molten salt reactor (MSR) is an emerging as a fourth-generation nuclear power reactor. MSR is a nuclear fission reactor where the reactor coolant and nuclear fuel are in homogeneous liquid state. MSR is a chemistry reactor and therefore, physical properties and chemical behavior of actinide and lanthanide elements is important in designing, operating and monitoring the process. Among many physical properties, viscosity is of our current interest, because it is a key heat transport properties in terms of thermal hydraulics.

Low-viscosity rotational viscometer is used to examine the accuracy of viscosity at temperatures 600–900 K. Temperature is a major factor in the determination of accuracy of viscosity measurement. Accurate viscosity measurement comes from the inaccurate temperature measurement, due to the temperature variation inside the vessels containing molten salt. In addition, actual temperature inside the vessel can be significantly different from the set temperature of the furnace. However, it is very difficult to measure the temperature inside the vessel since the gap between spindle and vessel wall is so small that the thermocouple cannot be placed inside the vessel in the rotational viscometer. The accuracy could be improved by the temperature calibration method along with cross-validation using potassium nitrate (KNO_3) as an internal reference material. In the temperature calibration method, the multiple thermocouples were placed into the molten salt horizontal and vertical temperature variation were predetermined, and the information on the temperature distribution will be used later for the temperature correction. Temperature correction was further confirmed by comparing the reference viscosity value of KNO_3 from the reliable report for the cross-validation.

Keywords: Molten Salt, Viscosity, Potassium nitrate, Temperature correction, Cross-validation

Fabrication of the Thermal Convection Loop for Multi-purpose Molten Salt Experiment

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Generally, the nuclear reactor used solid fuel and designed to use coolant for heat transfer. But the molten salt reactor (MSR) can be designed to use liquid fuel as a coolant. In MSR, liquid fuel uses molten salt, usually a eutectic mixture with a low melting point and a high boiling point. At high temperatures, molten salt has heat transfer properties similar to those of water. Molten salt has a strong tendency to corrode materials and its properties are easily changed because it reacts sensitively to oxygen and moisture. Therefore, molten salt to be designed to control disadvantages and tighten advantages. In the reactor, molten salt circulates in an airtight structure with a metallic material and is exposed to nuclear fission reactions, corrosion reactions, and transmutation reactions, producing various by-products. They continue to accumulate and changed the component and composition of molten salt fuel over time. The Associated thermal physics and thermal chemicals properties can be changed. In this study, review of previous researches to design the multi-purpose molten salt loop for the corrosion experiment of structural materials, the test of online component analysis, and the test of virtual fission product removal. On that basis, design requirements established and suggest the concepts of the multi-purpose molten salt loop.

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Keywords: Molten salt reactor, Molten salt, Liquid fuel, Thermal convection loop

Study of RCS CRUD Thickness Evaluation Methodology in PWRs

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CRUD (chalk river unidentified deposition) is generated by corrosion and erosion of the out-core surface, it is deposited on the surface of the fuel rod in the in-core. The deposited CRUD is activated by neutron irradiation and released into the out-core. CRUD is deposited on the RCS (reactor coolant system) surface, contributing to an increase in the exposure dose of workers. In order to prevent such problems in advance, it is necessary to analyze the nuclide concentration and element composition of CRUD. In this study, the methodology is suggested to estimate the deposited CRUD thickness using radioactivity. This methodology is considered with the empirical value in NPPs and the calculated value using the water chemistry variable. If nuclide analysis is operated using the measured radioactivity, the mass of nuclide per unit area can be obtained as the specific radioactivity of each nuclide. And the deposition rate is calculated in the computer code using water chemistry variables. Using the mass of the nuclide and the deposition rate, the CRUD thickness can be estimated. The error occurring in the CRUD thickness estimation can be considered for the following reasons. First, the deposition rate is a variable theoretically calculated using the material of the pipe, the temperature, and pH of the coolant, and it may vary depending on the characteristics of each pipe. And the sample collected through the sampling line may have a difference in CRUD concentration and radioactivities depending on the location of the sampling line and the sampling method. And, in the case of measurement using a radiation detector, a difference of radioactivity may occur while representing a large area as a measurement value of a local area. The CRUD thickness evaluation methodology presented in this study can be used as a basic method for evaluating the CRUD thickness in the RCS of the target nuclear power plant. Future research plans to refine the thickness evaluation factors and draw appropriate correlations to improve the methodology for precise thickness evaluation.

Keywords: CRUD, RCS, Radioactive corrosion product, RCS sampling

Electrochemical Extraction of Lanthanide Ions to Liquid Bismuth in LiCl-KCl Molten Salt

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The liquid-liquid extraction process using molten salt-liquid metal is one of the techniques for separating lanthanides and actinides in the salt. This study was conducted to obtain the information behavior of lanthanide elements between molten salt (LiCl-KCl) and bismuth liquid metal electrode. A three-electrode system consists of a rotating solid tungsten electrode and a liquid bismuth electrode was designed and conducted to reduce dissolved lanthanide. Experiments used five lanthanide chlorides (LaCl_3 , CeCl_3 , PrCl_3 , NdCl_3 , SmCl_3). The working electrode used a 5 mm tungsten rod and rotates at a speed of 100rpm. The counter electrode was made by putting a 1 mm tungsten wire inserted into a 1 mm pyrex tube with a bismuth pool. As the reference electrode, an Ag/AgCl (1wt%) electrode was used by putting LiCl-KCl containing 1wt% AgCl and 1 mm silver wire in a closed-end 2 mm inner diameter glass tube. The electrochemical experiment was conducted in two cases, methods were chronoamperometry and chronopotentiometry. In case of chronoamperometry, the current was recorded for 4 hours at an applied potential of -1.5 V (reduction potential of the Ln-Bi intermetallic compounds). In the case of the chronopotentiometry, applied of -0.30 A/cm^2 for 4 hours, and the potential value was recorded according to time. The measured potential gradually decreased at 0.005 V and then decreased sharply from about 2,500 seconds, which was subsequently maintained at -1.7 V without reduction. In these results, -1.7 V appears to be a reduction potential of Ln-Bi intermetallic compounds. In future research, it is necessary to separate each part and perform ICP-MS analysis to confirm the constituent elements and composition. The measured potential gradually decreased at 0.005 V and then decreased sharply from about 2,500 seconds, which was subsequently maintained at -1.7 V without reduction. In these results, -1.7 V appears to be a reduction potential of Ln-Bi intermetallic compounds. As a result of visual confirmation after electrochemical reduction, it was confirmed that the color was different at the interface between the molten salt/liquid metal layer. Subsequent studies necessary to separate each part and perform ICP-MS analysis to confirm the constituent elements and composition.

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This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP: Ministry of Science, ICT and Future Planning) (No. 2020M2D4A1070724).

Keywords: Molten salt reactor, Electrochemical extraction, LiCl-KCl, Molten salt

Upgrading of Delayed Neutron Activation Analysis System in HANARO

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Delayed Neutron Activation Analysis (DNAA) system and Pneumatic Transfer tube System (PTS) system had already been installed and operated in High-flux Advanced Neutron Application Reactor (HANARO) facility at KAERI. The DNAA at HANARO had performed very well in the series of characterization tests performed for ^{235}U samples. DNAA consists of 18 He-3 proportional tubes, polyethylene moderator, and 10 cm thickness of lead for shielding gamma background. 18 He-3 tubes are placed in cone concentric ring around PTS flight tube and polyethylene moderator surrounds He-3 tubes and the flight tube. The data acquisition system (DAQ) has a high voltage supply that can provide 0 to 3,000 V and PDT 110A board including preamplifier (+12VDC), amplifier, discriminator, analog to digital converter (TTL output), and signal summing circuit. During the long term (~ 6 years) overhaul of HANARO reactor, because DNAA has been not operated. Currently, the HV and bias DC voltage cable connections has to be replaced. As HANARO reactor re-start ramping up, we are considering upgrading the detector so that new DAQ system read and save the signal from one each of 18 He-3 separately and new HV and DC connectors are introduced. In this study, we describe the status of DNAA system upgrade and the plan of DNAA application with new DAQ system.

Keywords: Delayed neutron, Activation analysis, HANARO, He-3

Concentration of Epsilon Particles in Spent Nuclear Fuel

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In the field of spent fuel management, research on the epsilon particles with their special behavior is getting attention. Epsilon particles are fission products of nuclear fuel and they consist of Ru, Mo, Pd, Tc and Rh in phase of noble metal. Similar to fission gas bubbles, they exist as spherical form in the grain boundary. Epsilon particles are potential cause of cracking in spent fuel due to that they induce defects in solid solution of uranium oxide pellet and change the lattice volume. However, possible advantages of epsilon particles are also known. Since they can immobilize volatile elements such as Tc, radioactive release during fuel failures can be reduced. Also, oxidation reaction of uranium oxides can be decelerated due to galvanic reduction of epsilon particles to uranium.

In order to conduct a research related to epsilon particles, it is important to setup the accurate conditions with quantified concentration of five metal elements. The first method of predicting the concentration of the elements is computational modelling. SCALE/ORIGEN-6.2 code was used to calculate fuel depletion and cooling. Radioactivity and weight concentration of five metal element in spent fuel was evaluated. The other method is based on experimental method. OECD/NEA provides SFCOMPO-2.0 (Spent Fuel Isotopic Composition), the database composed of destructive analysis data of spent fuel from various reactors. The data of uranium dioxide fuel from pressurized water reactors was chosen for statistical analysis. From the experimental report enclosed in the database, the five elements from insoluble residue were treated with further steps with strong acid treatment and fusion. However, the results of residue solution and spent fuel solution were combined together.

Epsilon particles have potential impact on safety of spent fuel management. It is important to evaluate the accurate concentration of epsilon particles in spent fuel and setup the proper condition to proceed further studies.

Keywords: Spent fuel, Metallic precipitate, Epsilon particles, High-level waste

CuAF₆-44'bipyridine (A= Zr and Hf) Metal–Organic Frameworks as Promising Adsorbents for Kr/Xe Capture

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Separation of xenon (Xe) and krypton (Kr) plays an essential role in radioactive waste management, especially removing radioactive Kr-85 is an important step in nuclear fuel waste treatment. Due to the low concentration of Kr and Xe in the atmosphere, conventional method to separate these two gases is cryogenic installation. The method is focused on isolation of xenon and krypton using a cryogenic distillation, which greatly complicates the process and increases energy costs. Nevertheless, even after using the cryogenic distillation, concentration of radioactive Kr in the Xe is still high.

Metal–organic frameworks (MOFs) are porous crystals which have diverse pore geometry and unique physicochemical properties, and therefore they are successfully used for separation, adsorption and storage. The ability of MOFs to adsorb gases at room temperature can significantly improve the cost-effectiveness of the gas purification process. As it is well-known, copper complexes could be used in MOF engineering. Combination of framework-builder (Cu (II) ion and 4,4'-bipyridine (4,4'-bpy) ligand) and framework-regulator (AF₆ type anions; A= Zr and Hf) provides a series of new porous coordination frameworks, which can be applied for Kr/Xe separation.

This work describes the synthesis and characterization of new three dimensional metal-organic framework based on 44'bipyridine/copper (II) hexafluorozirconate as a potential adsorbent for Kr and Xe capture. The crystal structure of CuAF₆-44'bpy (A= Zr and Hf) was characterized using powder X-ray diffraction (PXRD) and single crystal X-ray diffraction (SC-XRD).

Keywords: Kr/Xe capture, Gas adsorption, Metal–organic frameworks, Single crystal X-ray diffraction

Synthesis and Characterization of Prussian Blue Analogue and Iron Oxide Composite Using XAS and XMCD

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For the selective removal of cesium ion, we synthesized nickel Prussian blue/maghemite (NiPB/ γ -Fe₂O₃) composite materials. NiPB/ γ -Fe₂O₃ composite was synthesized via a simple hydrothermal method. NiPB was dispersed by ultrasonication in deionized water, and the well-dispersed one was transferred into a Teflon-lined stainless-steel autoclave. The autoclave was then sealed, heated at 140°C for 4 h, and cooled to room temperature. The resultant NiPB/ γ -Fe₂O₃ has magnetic properties, which allow it to be separated from the solution contaminated by cesium.

We also measured to confirm its origin of magnetism by X-ray absorption spectroscopy (XAS) and X-ray magnetic circular dichroism (XMCD) measurement. From the XAS measurements, the Fe *L*-edge spectra of sample show characteristic peaks that can identify the valence states. The peak at 712 eV induced by a strong metal-ligand charge transfer in Fe²⁺ (*t*_{2g}⁶) decreases and the peak at 707 eV originated from a *t*_{2g} hole in Fe³⁺ (*t*_{2g}⁵) increases after the hydrothermal treatment, which means that the hydrothermal reaction. In addition, a peak emerges at 708 eV and the peak at 710 eV shifts to low photon energy. These changes indicate that the partial oxidation induces the somewhat iron oxides. In the XMCD measurements, the MCD spectra give us a typical MCD spectrum of γ -Fe₂O₃ where two peaks near 708 eV and 710 eV are attributed to the Fe³⁺ in O_h sites and an opposite peak near 709 eV comes from Fe³⁺ in T_d site. Although the oxide content is relatively smaller than the NiPB, the XMCD measurement clearly shows that the origin of magnetism is from γ -Fe₂O₃.

Keywords: Prussian blue analogue/ γ -Fe₂O₃ composite, Magnetism, XAS, XMCD

Effect of Humidity on the Electrical Conductivity of Zr- and Gd- Doped UO_2

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The UO_2 fuel irradiated in the nuclear reactor contains various elements, such as fission products generated during irradiation and some elements from cladding materials. Those elements may react each other to form a solid solution. Although UO_2 itself has Mott insulator characteristics, the electrical conductivity can increase significantly with a small amount of dopants. We used ZrO_2 and Gd_2O_3 as dopants which can form the solid solution with UO_2 to investigate the effect of the dopant level on the electrical conductivity. Furthermore, we also investigated the effect of humidity on the electrical conductivity.

Various compositions of $\text{U}_{1-x-y}\text{Zr}_x\text{Gd}_y\text{O}_{2+\delta}$ ($0 \leq x \leq 0.1$ and $0 \leq y \leq 0.01$) were prepared, and the electrical conductivity was measured by 4-point probe method, and absolute humidity between $7.08 \text{ g}\cdot\text{cm}^{-3}$ (dew point 6.6°C) and $14.03 \text{ g}\cdot\text{cm}^{-3}$ (dew point 16.9°C). At the humidity of $14.03 \text{ g}\cdot\text{cm}^{-3}$, at $x = 0.05$, the electrical conductivities increase from 0.10 to 1.51 S/m as y increases from 0 to 0.01. At $x = 0.1$, it increases from 0.05 to 0.35 S/m as y increases from 0 to 0.005. At fixed $y = 0.005$, the electrical conductivities decrease from 1.64 to 0.35 S/m as x increased from 0 to 0.1. And at $y = 0.01$, it decreases from 2.11 to 1.51 as x increased from 0 to 0.05. As the humidity decreases from 14.03 to $7.08 \text{ g}\cdot\text{cm}^{-3}$, the electrical conductivity appears to increase from 1.11 to 1.29 S/m at $x = 0.05$ and $y = 0.005$, whereas it increases from 1.51 to 1.70 S/m at $x = 0.05$ and $y = 0.01$. At $x = 0.1$ and $y = 0.005$, the electrical conductivities were 0.35 and 0.37 S/m at the humidity 14.03 and $7.08 \text{ g}\cdot\text{cm}^{-3}$, respectively. In conclusion, that electrical conductivity does not depend on the humidity significantly. Further studies are in progress to examine the electrical conductivity more accurately under the well-controlled environments.

Keywords: Doped uranium oxide, Zirconium, Gadolinium, Electrical conductivity, Humidity

Surface Characterization of ϵ -particle Doped Uranium Dioxide

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Microstructure and components of spent nuclear fuel are different from initial fresh UO_2 fuel before irradiation. Because of formation of fission products, transuranium elements and activation products, physical and chemical properties of UO_2 matrix were changed. ϵ -particle doping on fuel could cause the distortion of UO_2 cubic matrix and change of the chemical reactivity. These effects would provide the surface activation site that could enhance chemical reactions and corrosion processes. Therefore, it is important to characterize the surface structures and properties of ϵ -particle doped fuel when the final strategy for the disposal of SNF is considered with the failure case of barrier.

ϵ -particle doped uranium dioxide was manufactured using solid reaction process. Calculated amounts of UO_2 and ϵ -particle oxide powders were weighed and blended. The powder mixtures were pelletized by compressing. The sample pellets were sintered in the tube furnace at 1,700°C for 18 h with hydrogen atmosphere. The sintered pellets were cooled to room temperature after annealing while maintaining hydrogen atmosphere at 1,200°C for 12 h. X-ray diffraction (XRD) method was applied to obtain lattice parameter of sample pellets. Bruker D8 Advance using $\text{CuK}\alpha$ line source filtered with a Ni foil (beam current 40 mA at 40 kV) was used to take XRD spectra. The lattice parameters of the samples were calculated from XRD spectra with refinement process. The surface morphologies of sample pellets were measured by scanning electron microscopy (SEM). SEM images were obtained using electron acceleration of 20 keV. Raman spectroscopy was used to characterize the surface structure of each pellet. Raman spectra were obtained using ANDOR Shamrock SR500i spectrometer with a wavelength of 632.8 nm from He-Ne laser.

The surface structures of various ϵ -particle doped uranium dioxide instead of spent nuclear fuel were characterized using XRD, Raman spectroscopy and SEM. The influences of ϵ -particle doping on the surface structure of UO_2 matrix are compared and discussed.

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Keywords: Uranium dioxide, ϵ -particle, Surface structure, Structure characterization

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