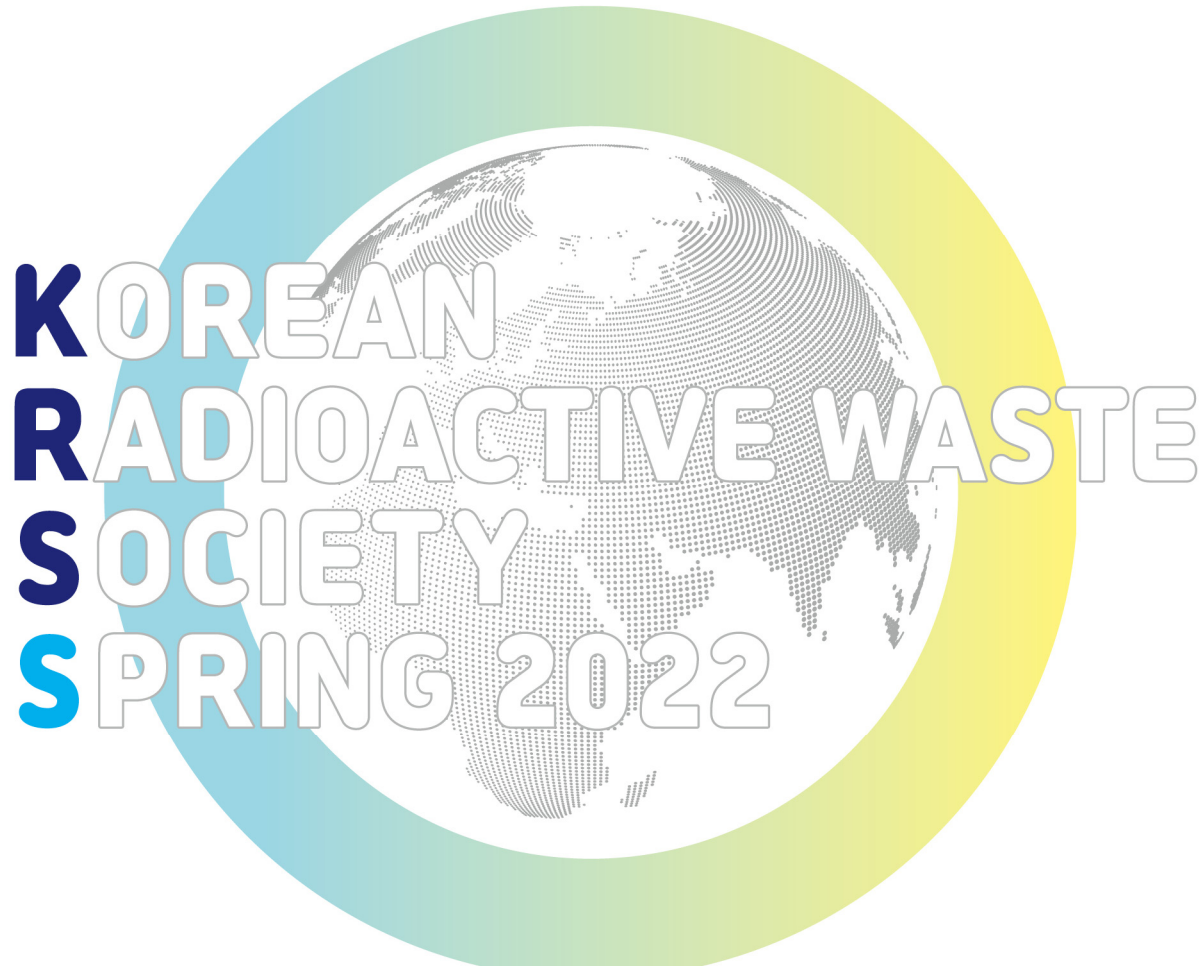


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**KOREAN
RADIOACTIVE WASTE
SOCIETY
SPRING 2022**



사단 **한국방사성폐기물학회**
법인 Korean Radioactive Waste Society

주최/주관

한국방사성폐기물학회

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1분과

핵주기정책·규제 및 비확산(Oral)

Current Status and Future Prospects on Nuclear Forensics Technical Capability Buildings at the Integrated Support Center for Nuclear Non-proliferation and Nuclear Security at the Japan Atomic Energy Agency

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Nuclear security event involving nuclear and other radioactive materials outside of regulatory control (MORC) has the potential to cause severe consequences for public health, the environment, the economy and society. Each state has a responsibility to develop national nuclear security measures including nuclear forensics to respond to such events. In Japan, national nuclear forensics capability building efforts mainly based on research and development (R&D) have been conducted since 2010, in accordance with national statement of Japan at the Nuclear Security Summit in Washington DC. Most of that work is undertaken at the Integrated Support Center for Nuclear Non-proliferation and Nuclear Security (ISCN) of the Japan Atomic Energy Agency (JAEA) in close cooperation with other competent authorities. The ISCN has made increased contributions to the enhancement of international nuclear security by establishing technical capabilities in nuclear forensics and sharing the achievements with the international community.

The ISCN has mainly engaged in R&Ds for establishing and enhancing nuclear forensics technical capability. As for the laboratory capability, several new pieces of analytical equipment have been introduced for nuclear forensics R&D purposes. High-precise measurement techniques validated in the past nuclear forensics incidents have been established, and novel techniques that can contribute to the more timely and confident nuclear forensics signature analysis have been developed. The ISCN has been also developed a proto-type nuclear forensics library based on the data of nuclear materials possessed for past nuclear fuel cycle research in JAEA. These technical capability developments have been conducted based on the cooperation with international partners such as the U.S. Department of Energy and EC Joint Research Center, as well as participation in exercises organized by Nuclear Forensics International Technical Working Group (NF-ITWG).

Recent R&D works have been mainly based on the needs of domestic competent authorities, such as first responders and investigators, and aim to develop technologies covering the entire spectrum of nuclear forensics processes from crime scene investigation to laboratory analysis and interpretation. One important key issue is the enhancement of technical capability for post-dispersion nuclear forensics. For instance, the ISCN has carried out the development of radiation measurement equipment coupled with the low-cost and mobile radiation detectors that uses machine-learning algorithms for quick and autonomous radioisotope identification to support first responders during crime scene investigations. Laboratory measurement techniques for samples collected at a post-dispersion crime scene are also among the important technical issues studied at the ISCN. The application of emerging technologies to nuclear forensics has also been studied. This includes the application of deep learning models to nuclear forensics signature interpretation that could provide more confident results, as well as the development of contamination imaging technology that could contribute to the analytical planning on the samples in collaboration with conventional forensics.

Many analytical techniques have been developed and the capability to analyze nuclear and other radioactive materials for nuclear forensics purposes has been considerably matured over the past decade. The challenges of post-dispersion samples, collaboration with conventional forensics and the development of novel signatures will be more important in the near future. Therefore, the ISCN will promote the R&Ds to further enhance the technical capabilities solving these issues. In addition, the ISCN is also promoting to expand the nuclear forensics research into universities and other research institutes in Japan. This is expected to contribute to the establishment of a domestic nuclear forensics network that enables to respond timely and flexibly to the MORC incidents, and to the maturation of nuclear forensics as a new academic field.

Keywords: Nuclear forensics, Nuclear Security, Nuclear material assay, Radiation measurement, Microscopic analysis

Tritium Productivity Evaluation of Graphite-moderated Reactor Using MCNP6

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North Korea claimed to have tested a hydrogen bomb in its fourth nuclear test in 2016, and declared that the hydrogen bomb was completed after the sixth nuclear test in 2017. North Korea's operation of Yongbyon Graphite-moderated reactor has been thought to be aimed at producing plutonium, but it has been strongly argued that the restart of the Graphite-moderated reactor is, indeed, aimed at supplying tritium recently. Tritium can be used not only to manufacture hydrogen bombs, but also to miniaturize nuclear weapons, making it as a key material for nuclear weapon capability. Since upgrading nuclear weapons and developing hydrogen bombs through the use of tritium by North Korea could pose a major threat to the security of the Korean Peninsula, it is important to accurately evaluate North Korea's nuclear weapon capabilities through the analysis of nuclear material production scenarios based on its nuclear facilities. However, researches on North Korea's nuclear materials such as HEU (Highly Enriched Uranium) and Pu production has been actively conducted, while no research has been shown on tritium production yet. Therefore, this study aims to evaluate the tritium productivity based on the analysis of hypothetical nuclear material production facilities and possible tritium production scenarios. Basic research was conducted about the existing theoretical methodology for tritium production, the analysis of the global tritium production history, and the analysis of nuclear facilities. Based on this basic investigation, feasible tritium production scenarios were constructed. Subsequently, based on design criteria of an hypothetical Graphite-moderated reactor, possible tritium production scenario was modeled by applying the TPBAR (Tritium Production Burnable Absorber Rod). In addition, the factors such as ⁶Li concentration, design parameters, material compositions, and the number of TPBARs, which may affect tritium throughput were analyzed in terms of sensitivity study such that the maximum and minimum throughput can be predicted.

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Keywords: Tritium, Nonproliferation, Nuclear safeguards, Nuclear forensic, Nuclear material accountancy

Performance Evaluation of In Situ Object Counting System (ISOCS) for Uranium Measurement in Radioactive Waste Drums

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For the peaceful use of nuclear energy, the international community has devoted itself to fulfilling its obligations under the Safeguards Agreement with IAEA. In this regard, uranium in a radioactive waste drum should be analyzed and reported in terms of mass and ²³⁵U enrichment. In order to characterize radioactive wastes, gamma spectroscopy techniques can be effectively applied. In the case of high-resolution gamma spectroscopy, because an HPGe detector can provide excellent energy resolution, it can be applied to analyze a mixture having a complicated isotopic composition. However, other substances such as wood, concrete, and ash are mixed in radioactive waste with various form factors; hence, the efficiency calibration is difficult. On the other hand, In Situ Object Counting System (ISOCS) has a capability of efficiency calibration without standard materials, making it possible to analyze complex radioactive wastes. In this study, the analysis procedure with the ISOCS was optimized for quantification of radioactive waste. To this end, a standard radioactive waste drum at KEPCO NF and low-level radioactive waste drums at Korea Radioactive Waste Agency (KORAD) were measured. The performance of the ISOCS was then evaluated by Monte Carlo simulations, Multi-Group Analysis for Uranium (MGAU) code, and destructive analysis. As a result, the ISOCS showed good performance in the quantification of uranium for a drum with the homogenized simple geometry and long measurement time. It is confirmed that the ISOCS gamma spectroscopy technique could be used for control and accountancy of nuclear materials contained in a radioactive waste drum.

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Keywords: Safeguards, Gamma Spectroscopy, ISOCS, Nuclear material, Radioactive waste, MCNP

A Study on the Comparison of Import and Export Procedures of Items Subject to Nuclear Cooperation Agreement Between the United States, Canada and Australia

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The Nuclear Safety Act defines items defined in nuclear cooperation agreements with each country as internationally regulated materials and requires that import and export procedures be implemented according to each agreement. In particular, the US, Canada, and Australia, which are major nuclear power suppliers, describe detailed procedures related to imports and exports in administrative agreements attached to the agreement. This paper compares and analyzes the import and export procedures of agreed items in the three major countries and proposes procedures and precautions that nuclear companies should follow for smooth import and export. The import/export procedure, according to the annexed administrative agreement, is divided mainly into Direct-transfer, Indirect-transfer, and Re-transfer. Direct transfer refers to direct import and export between countries that have signed an agreement. The four-step process of prior notice, written confirmation, shipment notice, and receipt confirmation is commonly required for direct transfer. However, in the case of Canada, if the previously related information is the same, the shipping notification and receipt confirmation procedure may be omitted. Australia only defines items included in the notification without a separate form in the administrative agreement. Indirect transfer means transferring items originating from the other two countries that signed the agreement through a third country. All three countries stipulate that item transferred by indirect transfer are also subject to the agreement, and Canada stipulates separate advance notifications and shipping notifications for indirect transfer. Australia manages indirect transfers by including information from third countries in the exchange-related information between the two countries. The US does not have a specific procedure for indirect transfer, but it is presumed that it was omitted because it was difficult to confirm the time of shipment when the US exports through a third country. Re-transfer is a procedure to obtain consent from the original exporting country in advance when exporting items held in Korea to a third country. There are two types of consent based on re-transfer, the long-term consent method, and the individual case-by-case method. Long-term consent is a method of long-term consent for re-transfer to an agreed-upon country by agreeing in advance on a list of countries where re-transfer is possible. In the long term, the procedure will be reflected in domestic laws in detail and managed through an IT-based management system so that operators can smoothly implement such complex import and export procedures.

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Keywords: Nuclear cooperation agreement, Administrative agreement, Direct-transfer, Indirect-transfer, Re-transfer

Development of Data-communication Technology for Real-time Active Seal

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In order to enhance the transparency of nuclear power, the investigation of nuclear activities is important at the right time. And investigation of North Korea, one of safety measures for such a surveillance system seems essential in the near future. However, the inspection period is limited and the possibility of revisiting is unclear, so it seems easy to install and more reliable monitoring devices are needed. The seal is a security device for monitoring forgery and alteration in the IAEA easily but simple. Since the existing seal devices used by IAEA and KINAC can only be checked through on-site inspections whether it is damaged or not, there is a limit to immediate response in the event of an emergency such as the theft of nuclear materials and loss of Continuity of Knowledge. While the North Korea-U.S. talks and inter-Korean talks are slowly approaching the North Korea's nuclear issue, it is judged that the inspection of North Korea's nuclear weapons is not in the distant future. Accordingly, it is necessary to develop a customized seal device for managing and monitoring spent nuclear fuel that can be used for inspection of North Korea. The real-time active seal device consists of a seal equipped with a detection sensor, a translator, and a server, and the translator use wireless communication in consideration of ease of installation and management, respectively. The translator and server use wired communication for data sustainability. The seal device is designed to maintain continuity of knowledge (COK) through wireless communication in consideration of security even in the event of equipment damage, malfunction, and loss of power, while enabling location detection in case of theft or abnormality. Therefore, in this study, a technical investigation on remote data communication was conducted accordingly, and a data communication module for a small amount of broadband data communication was manufactured and applied. In addition, several tests were conducted to determine data integrity and location accuracy according to the communication range.

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Keywords: Seal, Active, Real-time, Data-communication technology

Development of the Diversion Path Generation Algorithm to Apply to the Safeguards Analysis

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In order to effectively and efficiently apply safeguards to new nuclear facilities, it is recommended to apply safeguards-by-design concept. In evaluating the safeguards in the early stage of the design of a facility, it is essential to analyze the diversion path for nuclear materials.

This study suggests a simple method which can generate diversion paths. The essential components constituting the diversion path were reviewed and the logical flow for systematically creating the diversion path was developed. The path generation algorithm is based on this components and logical flow as well as the initial information of the nuclear materials and material flows in a planned facilities.

The results will be used to develop a program module which can systematically generate diversion paths using the event tree and fault tree method.

Acknowledgements

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

A Strategic Adoption Model for Blockchain Technology in Nuclear Nonproliferation

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This paper presents a strategic adoption model for blockchain technology in nuclear nonproliferation by analysing the implementation of legally binding agreements and leveraging results from government-leading sectors. Blockchain has been emerging as not only a single promising technology but a foundational one which can be combined with diverse sectors. From the national point of view, it is imperative that the government formulates policy for fostering blockchain-related industries, thereby, gaining a competitive advantage at the national level. Accordingly, the Korean government has established the Blockchain Technology Development and Diffusion Strategies in 2018 and 2020, respectively, to verify the technology by supporting pilot projects for apposite industries, such as customs clearance, transshipment of containers, record-keeping of meat processing, and smart contracts. In addition, the strategies announce to support liaison with regulatory sandbox and cooperation between the projects. Internationally, on the other hand, nuclear nonproliferation imposes the duties of verifying that member states under the NPT and the Safeguards Agreement obey the IAEA mandate, “Atoms for Peace and Development”. Similarly, bilateral nuclear cooperation agreements and administrative arrangements specify reporting obligations for the origin and history records of the Trigger List items. Meanwhile, commercial and industrial secrets and other confidential information of any entities involved have to be securely protected. Provided accompanying activities accomplish the integrity of records and mutual transparency, it brings more credibility, and further, the competitiveness of the state’s nuclear industry. In conclusion, the tasks that the Republic of Korea implements as an exemplary country complying with the nuclear nonproliferation regime have many similarities with the pilot projects that have been or are being carried out under national strategies for fostering blockchain technology elsewhere. This implies that the leveraging of the subsets can derive a new competitive model in blockchain adoption that contributes to the competitiveness of the national nuclear industry due to the advanced nuclear regulations.

Keywords: Nuclear nonproliferation, Blockchain, Strategy, Leveraging

1분과

핵주기정책·규제 및 비확산(Poster)

Development of Installation Standards and Evaluation Indicators for CCTV Systems in National Security Facilities

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In December 26, 2017, the Ministry of Science and ICT's Cyber Infringement Response Division, the KCC's Internet Ethics Team, and the National Police Agency's Cyber Investigation Division announced the "Comprehensive IP Camera Measures" to prevent damage to users by strengthening IP camera security. The background was a countermeasure against the spread of public anxiety such as leakage of privacy and threatening national security as cases of illegal filming and distribution of videos by accessing IP cameras without permission at that time occurred. At that time, the measures consisted of three major strategies: institutionalization of products with security (manufacturing and import stage), rapid action and response to hacking threats in advance, and intelligent advancement of IP cameras (industrial development) and fostering various video and safety industries. This paper deals with the development of installation standards and evaluation indicators for CCTV systems installed in the relevant national security facility. When designing a CCTV system, the resolution and sensitivity of the camera, the angle of view of the lens, and the aperture should be selected in consideration of the length and width of the monitoring area. When it is necessary to determine an alarm based on the recorded image information, the resolution of the recording device should also be considered. If there is a restriction on the surveillance area of the camera due to topographic features, visitors, etc., additional cameras should be considered. Additional surveillance cameras should be considered to effectively monitor sections where fences are bent or sections with severe slopes. When designing a CCTV system, supplementary means that can be used for intrusion monitoring should be devised even if the entire or part of the system causes a failure. KISA's performance test and certification procedures shall apply mutatis mutandis to intelligent CCTV image analysis test procedures and certification. Intelligent CCTV image analysis evaluates the performance of the equipment by comparing files of ground truth (GT) and system alarm (SA) for verification to calculate the evaluation score.

Keywords: CCTV systems, National security facilities, Installation standards, Evaluation indicator

Development of “VR Shooting Training Center” to Enhance the Field Applicability of Physical Protection Education

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In the event of an contingency situation of physical protection in nuclear facilities, the first organization to deal with at the forefront is the Special Response Forces (SRF). Since the SRF has to perform nuclear facility protection at the actual battle site, they must repeatedly train tactical understanding such as shooting, entry, and suppression so that their body can remember it even in real contingency situations (called Muscle Memory). In reality, however, repeated training using firearms is very difficult due to high risk and high cost, except for some military and police organizations. Using the advantages of VR technology, the Korea Institute of Nuclear nonproliferation and control (KINAC) has developed educational contents of “VR Shooting Training Center (VR STC)” to enable low-risk, low-cost, and repeated shooting training for these high-risk, high-cost training.

This content was developed by dividing it into an “indoor” and “outdoor” training field. Educational firearms are all developed as gas guns to add to the sense of reality, and trainees can choose firearms, distance movement of targets and other options.

The “Indoor training field” was developed by imitating an actual indoor shooting field, in particular the “outdoor training field” was developed using VR technology and motion tracking technology. Therefore, in “outdoor training field”, trainees can move freely within the designated spot of not only in VR content but also reality and then have to perform some missions. Trainees have to overcome random obstacles as they move to a designated destination, and at the destination, they are attacked by terrorists. Therefore, trainees must complete missions by concealing their bodies using objects around them. The one training course includes a total of 10 missions, and after the training is completed, comprehensive training results such as shooting accuracy and mission completion are expressed.

VR STC will be a representative example of making high-risk, high-cost training into low-risk, low-cost, and repeated training. In this respect, VR technology can be used to develop various radiation protection curriculums accompanied by high risk and high cost, and can improve educational effects.

Keywords: Physical protection, Training, VR, Contingency situation, Shooting

Development of E-learning Modules to Support IAEA Online Training Course Resources Under COVID-19

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MSSP means Member State Support Program, which is the program of the member state supporting IAEA. To efficiently carry out the safeguards, IAEA receives the support of the member states through this program and carries out R&D for the nuclear material inspection equipment, etc. ROK has been supporting and contributing total 21 tasks to IAEA in-cash and in-kind. These tasks consist of training, safeguards approaches, Analysis Support and NWAL Coordination, information analysis, Safeguards Information Systems and System Usability. ROK support program that consisted mainly of hosting IAEA inspector training courses was not active in 2021 due to COVID-19. So, through the project called “online course development consultation”, we developed e-learning module (titled “Introduction to Comprehensive Safeguards Agreements following the model text of INFCIRC/153 (corr.) related to State reporting obligations”) to provide continuous support for IAEA’s non-face-to-face safeguards activities in the current pandemic situation. We developed this module that is HTML5+MP4-based SCORM type to provide services for both PCs and smart phones. In conclusion, The ROK has differed in its form of support from previous years by developing e-learning modules for IAEA trainings and thus contributed towards IAEA’s international collaborative system to enhance its safeguards capability and this has enhanced the ROK’s status in the IAEA.

Acknowledgements

This development was supported by Nuclear Safety and Security Commission (NSSC).

Keywords: IAEA, MSSP, Safeguards, COVID-19, E-learning module

A Study on Safeguards by Design (SBD) for Small Modular Reactors in ROK

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IAEA, in preparation for possible commercial operation of small modular reactors (SMR), is pursuing the early development of safeguards approaches for these reactors along with the publication of the safeguards technical report (STR) to make verification activities easier for inspectors. For this purpose, the IAEA, through the MSSP, is developing various approaches for the application of safeguards by design for SMRs in collaboration with five countries that along with the ROK includes Russia, China, and U.S. etc. In order to review the specific design information of the ROK's only SMR facility, SMART, and to establish safeguards methods from the initial design stage, collaborations were made with the Korea Atomic Energy Research Institute, which researched and developed the SMART. As a result, the design information questionnaire (DIQ) and STR was created and sent to the IAEA. The DIQ is a collection of crucial questions regarding the application of safeguards to understand the characteristics of the reactor facility, and STR is a document referred to by IAEA inspectors during safeguards verification activities. The main contents of the STR consist of introduction, technological description of SMR, safeguards approach, conclusions and annexes. Through this study, it is anticipated to understand the technical requirements for safeguards implementation of SMRs in the design stage, and through the completion of the final report applying SBD with regards to the design of the SMART facility, it could be used as information material for future safeguard verification activities by IAEA inspectors.

Keywords: IAEA, Safeguard by design, Small modular reactor, DIQ, STR

A Study on the Necessity of Establishing an History Management System for Import and Export Items Subject to 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 US signed the Administrative Agreement (AA) under the new Nuclear Cooperation Agreement. Accordingly, when Korea imports items subject to the Korea-US agreement, it is required to determine the inventory of imported agreement items and notify the United States of the inventory amount every year. In addition, when re-exporting an agreement item to a third country, prior consent of the original exporting country (USA) must be obtained. Nuclear companies that import items subject to the Korea-US Atomic Energy Cooperation Agreement must report their inventory to the government every year, but the standards and procedures for managing the inventory are not clearly stipulated in the national law. This makes it difficult for the government to verify the adequacy of the report submitted by nuclear companies, adding to the administrative burden on both the government and the companies. Accordingly, it is required for the government to establish and operate a system for history management system for import and export items subject to the agreement so that related information can be recorded and managed at each stage, such as first import of items to Korea, generation, disposal, and exports to third countries. This system provides history management functions such as initial import information record for items imported through import/export procedures according to administrative agreements, change of owned company due to domestic movement, deletion of inventory due to loss/disposal, deletion of inventory due to export or addition of inventory due to derived materials. Through this system, operators can easily manage agreement items, and the government can obtain reliable information on agreement items in close to real-time. In addition, when this system applies to exports of items subject to the agreement, the number of items subject to the agreement exported by Korea can be provided first so that the importing country can more quickly check the items subject to the agreement. It is expected to contribute to securing control of the items subject to the agreement and reducing concerns over nuclear proliferation.

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Keywords: Nuclear cooperation agreement, Administrative agreement, Export-import, History management, Prior consent of retransfer

A Study on the Export Procedures and Management Methods for Spent Nuclear Fuel Subject to Nuclear Cooperation Agreement

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The ROK conducts several export procedures, communications in connection with transfers; exchange of information on export plan, shipments, and receipt of nuclear materials, in accordance with bilateral Nuclear Cooperation Agreements (NCA) and Administrative Arrangements (AA) signed with US, Canada, and Australia. Also, the inventory amount of items subject to NCA has reported annually. This study reviewed the export procedures and management methods for spent nuclear fuel subject to NCA.

The re-transfer procedures start with obtaining consent from the original exporting country. It is impossible to retransfer nuclear material without consent, whether long-term or individual case-by-case. If the material has multiple obligations, prior consent from all of those countries is required. Therefore, it is necessary to clarify the foreign obligated materials correctly.

In general, nuclear fuel is subject to multiple obligations of all countries through which the materials have passed during the front-end fuel cycle. Then the new obligations are imposed on those irradiated materials or their by-products after ‘used-in’ or ‘produced through the use of’ equipment subject to NCA. For example, fuel assemblies manufactured under CANDU fuel fabrication equipment subject to ROK-Canada NCA or burned in nuclear reactors where US equipment is installed have obligations based on Canada or US agreements.

In order to impose obligation to irradiated materials, the principle of proportionality is applied as stipulated in each Agreement. According to the AA between US and ROK, nuclear materials used in the equipment transferred under the Agreement and produced through them are differently controlled. After the cycle in the reactor with US-made equipment, uranium in the irradiated fuel is considered a material used in the equipment. So it would be appropriate to apply obligation proportionality according to its origin, regardless the US-made equipment. Meanwhile, the obligation under US NCA is given to the entire amount of produced plutonium in the irradiated fuel. Although the contribution to the production of fuel is to be discussed case-by-case basis in the case of Canadian obligation, applying a similar method is proper.

Since the fuel is burned in the form of bundles or assemblies, it is impossible to separate the spent fuel into uranium and plutonium physically. However, as discussed above, to clarify the rights and obligations pursuant to Agreement and ensure accuracy in inventory management, the obligation codes should be imposed on irradiated fuel as not a single item but separated individual substance of materials. Moreover, when an obligation swap occurs for the irradiated fuel, its movement and combustion history should be considered to prevent confusion in confirming multiple obligations and implementing export procedure.

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Keywords: Nuclear cooperation agreement, Export and import procedures, Spent nuclear fuel, Foreign obligation

Review on the Strengthening Export Controls in Response to Russia-Ukraine War

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After the annexation of Crimea in 2014, Russia continued to deploy military forces and equipment near the Ukrainian border in March and October of 2021, heightening the international crisis. On February 24, 2022, Russia began its full-scale invasion of Kyiv, the capital of Ukraine, with missiles and ground forces. Russia's invasion of Ukraine was accompanied by an urgent speech by Russian President Vladimir Putin on the day he would conduct a special military operation in Ukraine. Putin warned that Russia would seek to demilitarize Ukraine and retaliate immediately if foreign interference occurred. In particular, he stated that the expansion of the North Atlantic Treaty Organization and exploitation of Ukrainian territory was unacceptable. Due to the current Russian invasion of Ukraine, the United States has updated export controls and sanctions as of March 15. Extensive US export controls and sanctions recently imposed on some areas of Russia, Belarus, and Ukraine have included industries such as defense, aerospace, energy, and finance. Executive Order EO14065 is issued to ban transactions with specific individuals and entities, including financial institutions. Additionally, Executive Order EO14066 has banned US imports of Russian oil, natural gas, and coal and new investments in the Russian energy sector. The EAR was revised to strengthen export controls on Russia and Belarus. The sanctions imposed include 48 major defense companies, 328 personnel, and the CEO of Sberbank, which produced weapons used in the attack on Ukraine. Companies are listed on the Sectoral Sanctions Identification (SSI) List, and individuals are listed on the Specially Designated Nationals (SDN) List. Sanctions such as asset freezing and a ban on all financial transactions with Americans apply. In line with the international trend, Korea also declared its participation in sanctions against Russia. As of March 25, 2022, export controls have been strengthened by newly established items subject to catch-all licenses related to Russia and Belarus. Ministry of Trade, Industry, and Energy (MOTIE) added Fifty-seven items to Annex 2-2 in the Notification of Export and Import of Strategic Items. Most of these sanctions are for dual-use items under the jurisdiction of MOTIE. However, as countries, organizations, and individuals who may be subject to catch-all licenses are included in the sanctions list, Nuclear Safety and Security Commission should also review catch-all licenses for Trigger List Items. These sanctions are expected to last for some time. Even though China and Russia are the Nuclear Weapon States, the US has strengthened export controls. This is likely due to the opacity of China and Russia's export controls system and the lack of active implementation of UN Security Council sanctions. However, there is an aspect of protecting their technology. It seems that Korea should also pay attention to these changes in international trends and keep pace with the level of control in other countries.

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Keywords: Export control, Sanction, Catch-all license, Russia-Ukraine war

Application of Bayesian Network for Risk Assessment of Export License

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Efforts for nuclear non-proliferation have continued since the development of nuclear weapons and the conclusion of the NPT Treaty. Nuclear proliferation requires materials, facilities, and human resources to make nuclear weapons, and it takes a medium to long-term time. There are many restrictions in the current system to obtain nuclear materials and facilities, so it is often done through illegal means, black markets, or confidential transactions. Methods have been developed to evaluate the nuclear non-proliferation regime to strengthen the non-proliferation and solve the problems. The IAEA and the United States DOE initiated the proliferation resistance evaluation in 1980. The DOE conducted the assessment in three main evaluation categories: materials, technical characteristics of facilities, and institutional barriers. In another nuclear non-proliferation evaluation study, some researchers evaluated three main types: current capacity, political situation, and international situation. Detailed indicators include economic capacity, industrial capacity, nuclear capacity, leader's intentions, political structure, competitive relations, alliances, and international norms. Most of these evaluations are based on the situation at the time of assessment at the national level. Historical examples of nuclear proliferation are rare, and verification is also challenging. The Bayesian probability is widely used when the data is small, experiments are impossible, and the causal relationship is unclear. A Bayesian network is a combination of Bayesian probability and graphics. It is used throughout the industry because it can easily derive results according to causal relationships and weights of various variables, evaluate the risk for decision-making, and obtain changed results through data updates. In particular, to evaluate the proliferation of nuclear weapons, Freeman developed the Freeman network in 2008 and the Freeman-Mella network in 2014. Freeman explained in detail only the process of deriving variables, correlations, and probabilities of factors related to factors such as motivation, intention, and resources. It isn't easy to view as an objective result value because it does not describe the academic background for path selection, motivation list, intention, and resource variable selection. However, the research was meaningful because he first used the Bayesian network for nuclear proliferation. Although some studies have been done at the macro level, there is no case of applying it in export controls, which is the beginning of the actual spread. Also, there is no quantitative value for factors for risk assessment. There is little data, and verification of causality is difficult, so if the Bayesian network is applied to export control and applied to actual implementation, it will help make decisions such as export license or export denial.

Keywords: Export controls, Nuclear non-proliferation, Bayesian network

Conceptual Design of NPPs Cyber Security Test-bed

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The cyber-attack on Natanz nuclear facility in Iran which called Stuxnet showed how cyber could affect the physical system. If cyber-attack on NPPs compromise digital I&C system, it may occur some malfunction on actuators and at worst, radioactive material released into the environment. However, it is hard to test the cyber security on operating NPPs because of the safety problems. So, it is necessary to develop a test-bed to test both the cyber security of NPPs and the effect of cyber-attack on NPPs. KINAC has been developing NPPs test-bed to evaluate the cyber security of NPPs, validate cyber security controls of licensee and train the inspectors. In this paper, the conceptual design of NPPs cyber security test-bed will be discussed. Actual I&C systems such as PLC (Programmable Logic Controller) and DCS (Distributed Control System) are essential for testing cyber security. Also, NPPs simulator is one of important part to evaluate or analyze the effect of cyber-attack on NPPs. Usually, NPPs simulator consists of software which contains nuclear model, thermal-hydraulic model, execution program and GUI and hardware which contains workstation, operator console, PC and large display panel. It provides very similar to actual NPPs to users. However, in case of conventional NPPs simulator, I&C part is implemented as a software, so it is impossible to test the cyber security. To solve this issue, in case of the NPPs cyber security test-bed, I&C part should be hardware and simulation code should be modified to connect the hardware I&C part and software simulator using the HIL (Hardware-in-the-loop) method. The main purpose of this NPPs cyber security test-bed is to utilize in NPPs cyber security regulation. So, KINAC is developing the test-bed with APR 1400 simulator model and KNICS PLC and DCS platform. These real hardware I&C system will be connected to hacker's PC to test cyber security of NPPs. Also, the data set will be updated with real NPPs data set after the test-bed development finished. Furthermore, to give various analysis environment, archiving equipment that archive major plant process data, network packet between I&C systems and the like will be added. This NPPs cyber security test-bed combined the good points of conventional NPPs simulator and cyber security test-bed. It can test the cyber security of NPPs that conventional NPPs simulator cannot do. Also, it can evaluate and analyze the impact of cyber-attack on NPPs that cyber security test-bed cannot do.

Keywords: NPPs cyber security, Test-bed, HIL, Simulator, I&C system

A Study on Nuclear Legacy Site Management According to the International Management Guidance

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The nuclear facilities sites handling radioactive materials are contaminated due to accidents or its activities. Since the experimental reactor was operated in Idaho, USA in 1951, nuclear power and its related activities were carried out in many countries. Thus, the legacy sites in which radioactive materials have been deposited are a matter of public concern. As a result of the operation of facilities related to the use of nuclear energy, it is faced with the restoration of the environment containing radioactive wastes. In particular, the activities carried out in the early years caused its contamination of the sites and areas with significant releases of fission products and natural radioactive wastes adventitiously or intentionally due to the lack of its technology and the unclear radiation risks. In addition, nuclear weapons tests conducted in the 1950s and 1960s and the Chernobyl power plant accident in 1986 caused radioactive contamination in the Pacific Ocean and extensive territories in Europe. Accordingly, the IAEA discussed the restoration of the contaminated areas at the conference on the radiation legacy of the 20th Century in terms of environmental restoration (RADLEG-2000) held at Moscow, Russia in October 2000 and its cases from the former Soviet Union and Eastern Europe were reported. Also, the Nuclear Energy Agency (NEA) has discussed the potential issues on the legacy sites occurring radioactive materials and suggested that these sites should be managed in an open, transparent and consistent manner in order to build sustainable solutions with trust. However, there is currently no overall regulatory system for the management of legacy sites in Korea. The TRIGA Mark-II research reactor, which was commissioned in 1962, was decontaminated and dismantled in 1997. As such, it is necessary to consider the regulatory system related to the existing legacy site. Most of all, the legacy site management and regulatory standards of facilities that have operated in the past and are not currently operating or that deal with natural radioactive materials should be established. Also, the dismantling of NK's nuclear facilities and management of related radioactive waste, particularly NK's uranium mines and refining facilities particularly at Pyongsan and Bakcheon in NK are one of the key issue on the North Korea's denuclearization. In this study, the international standards on the reuse of the legacy site after dismantling are analyzed, and its regulatory considerations used for domestic application are suggested.

Keywords: Legacy site, Environment restoration, Radioactive waste, Nuclear regulation

Policy Suggestions for Spent Nuclear Fuel Management: Conformity Assessment System

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Since the commercial operation of Kori unit 1 in 1978, nuclear power has provided cheap, stable and clean electricity in South-Korea. For decades, the discussion about the spent fuel management has been dominated and the government is responsible for on-going research and development (R&D) related to long-term spent nuclear fuel management. The effective management of spent fuel should be applied from the early stage of the R&D process to licensing phases with the step-by-step evaluation system. As part of follow-up efforts after the Fukushima nuclear accident, the Nuclear Promotion Commission and Nuclear Safety Commission were divided in function as an independent agency for enhancing national nuclear safety and security, which aims to protect the public and environment from undue radiological hazard. The national spent fuel project must have a vibrant program for spent fuel management. Due to the nature of these projects, the establishment of a ‘conformity assessment’ system that collects the opinions from the licensing organisations on the results of research projects from the initial R&D stage should be applied advertently in order to efficiently conduct research projects and enhance public confidence. For the government-led project for spent nuclear fuel management, the adequacy and applicability of its technology R&D as well as its sustainability that includes financial, social and environmental performance measures should be evaluated in each stage. The institutionalisation arrangement, so called ‘conformity assessment system’ for the development of a national spent nuclear fuel management plan and related technology should be developed. This study aims to propose the basic principles for the introduction of the conformity assessment system: (1) national management responsibility, (2) spent fuel management project scope, (3) its management main principles, (4) project implementation system, (5) final management project scope and securing financial resources.

Keywords: Spent fuel management, Nuclear safety and security, Conformity assessment system

Proliferation Resistance Features for Molten SR Using ^{233}U - ^{232}Th Fuel Cycle

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Liquid-fueled Molten Salt Reactors (MSRs) do not contain their fuel in assemblies. It is then not possible to perform traditional item counting and visual accountability of the salt fuel. These facilities are closer to bulk accounting facilities, such as reprocessing plants, and require inventory determinations based on measurements of the actinide content of salts. This can be problematic due to the difficulty of sampling and the destructive analysis of actinide-containing molten salts. Some problems arise from the unique combination of high temperature and high radiation environments present in molten salt fuels. Another challenge is the continuous change in the isotopic concentration of fuel salts due to burn-up, conversion, plating out, and online chemical processing. There is a potential for fuel stocks outside the reactor containment vessel in on-site salt processing. In terms of proliferation resistance of ^{233}U - ^{232}Th fuel cycle, the nuclide ^{232}U is an important nuclide in thorium fuel cycle from the standpoint of proliferation resistance, because its daughter Thallium (^{208}Tl) is a strong gamma (2.6 MeV) emitter. The hard gamma ray is not only barrier from to nuclear material theft, but also an effective means of detecting lost fissile material. However, there is a theoretical weakness in obtaining pure ^{233}U at the core of the initial two weeks with a concentration of ^{232}Pu less than 1,000 ppm. Therefore, Pu separation process is one of the most sensitive parts in online reprocessing facility. The decision to use a fertile blanket should also be based on proliferation risk considerations in addition to operational parameters. MSRs can be designed without a separate fertile blanket, which should be considered. In the case of the MSFR, even if fertile blankets are used, the production of ^{232}U is large enough to make difficult the utilization of blankets for proliferation purpose. For the liquid-fueled MSRs without fissile materials separations, many of the observations from the previous section apply, except salt processing is minimized. The reactors will still need some method of estimating total actinide content. These reactor designs reduce proliferation risk for the reactor by not separating any actinides during operation.

Keywords: MSR, Fuel, Thorium, Proliferation resistance

The Difference for Proliferation Resistant and Physical Protection (PRPP) Concept for Liquid-Fueled Reactor

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Molten salt reactor (MSR) has a unique characteristic using liquid fuel and/or coolant salt among six type of GEN IV reactors. Liquid fuels and on-site processing are fundamentally different from a solid fuel reactor where separate facilities produce the fresh solid fuel and process the Spent Nuclear Fuel. Because the choice of fuel cycle affects the safeguards and non-proliferation characteristics of the reactor system, different MSR concepts may have different proliferation resistance and physical protection characteristics. For example, MSR design variants that use solid fuel but are cooled with liquid salts such as FHR are very close to the Very High Temperature Reactor design concept. The composition of various fuel salts is a representative factor that makes it difficult to generalize the PRPP evaluation principle of MSR. In addition, the flow of molten salts containing fissile materials is also complex depending on the design of the reactor. The path through which radioactive materials travel not only inside the reactor but also to nuclear fuel cycle facilities can act as a difficult factor in measuring nuclear materials. As a further complication, some of the plants include fuel salt drain tanks intended to provide decay heat removal while others are designed to provide decay heat removal while the salt is maintained within the reactor vessel. Some lessons learned from the prior molten salt breeder reactor program are reflected in all of the new designs. Interior reflectors/shielding are frequently employed to reduce the radiation damage to the reactor vessel, and fuel salt chemistry control is employed to substantially limit oxidizing the container alloy constituents. However, even with the vessel interior shielding, the containment environment around both solid and liquid fueled MSRs during operation is likely to have substantially higher dose rates than at LWRs due to the fission process and fission products in the case of circulating liquid fueled reactors, and the short-lived activation products of fluorine (^{16}N , ^{20}F , and ^{19}O) in the case of FHRs. Consequentially due to insufficient shielding from the coolant and the vessel wall, MSR containments will be remote access only for liquid fueled systems and remote access only during operation for FHRs.

Keywords: Molten salt reactor, Liquid fuels, Safeguards, Proliferation resistance and physical protection

A Study on Urgent Technology Transfer System for the License on Technology Export of Nuclear Plant and Current Issues

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The License on Technology Export of Nuclear Plant is a system that permits the export of strategic technologies for large-scale NPP projects collectively during the project period. So, an issuance of the export license could be omitted for each transfer of technology, but Post Strategic Items Confirmation must be performed before the transfer as a follow-up obligation.

Sometimes, transfers of technology have been urgently required during the project. As Post Strategic Items Confirmation process takes up to 15 working days, it may be difficult to respond to urgent situations timely, which may cause setbacks on the project. Therefore, Urgent Technology Transfer System, which allows to transfer technology without prior Post Strategic Items Confirmation, was established to reduce a burden on licensee and improve the efficiency of regulation. This system applies only to the License on Technology Export of Nuclear Plant. In other words, the technology transferred through Urgent Technology Transfer System (hereinafter referred as Urgent Transfer Technology) does not pose any problem with regard to export control because it is already licensed. In addition, the Urgent Transfer Technology should be considered as a strategic technology until Post Strategic Items Confirmation, which means that the Urgent Transfer Technology is more strictly controlled than the generally transferred technology. Also, the Urgent Technology Transfer System does not apply to intangible technology transfers such as technical support through personnel dispatch.

The system could be only used in specific conditions which are stipulated for each licensed project in advance in order to prevent indiscriminate abuse of the system by licensee. Licensees are required to report quarterly the stipulated condition corresponding to each Urgent Technology Transfer case, and it would be checked through post-site inspection whether the actual reason for the transfer meets the consulted condition. Moreover, the deadline of application on Post Strategic Items Confirmation after the Urgent Technology Transfer is stipulated for licensee so as not to omit the classification procedure.

This Urgent Technology Transfer System does not apply to dual-use items. If the Urgent Transfer Technology is classified as a non-Trigger List Item through the Post Strategic Items Confirmation, it is outside the scope of the NSSC's export license. In this case, the technology may be subject to an export license of the Ministry of Trade, Industry and Energy (MOTIE). However, if the technology is classified to be a dual-use item after Urgent Technology Transfer, it may result in unauthorized transfer because it has already been transferred. Licensee must apply to classification of MOTIE before Urgent Technology Transfer if the technology being transferred may be related with Dual-use Items. It is easy for licensee to overlook due to the low awareness about this system. Therefore, outreach activities are necessary to raise licensee's awareness by explaining the Urgent Technology Transfer System and current issues in detail. Consultation with MOTIE may be needed for the improvement on issues.

Keywords: Trigger list items, Dual-Use items, Urgent technology transfer system, Post strategic items confirmation, The License on technology export of nuclear plant

A Review of North Korea's Major Trading Partners and Items

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Investigating major trading partners and items with North Korea is informative in terms that it can predict the path through which North Korea's strategic items will transfer to non-nuclear-weapon states when North Korea denuclearizes. By analyzing North Korea's trading partners and the items, it is possible to identify the relevant countries through which items arrive from the first importing country to the end-user in the process of exporting items and to predict the way how North Korea disguise or conceal their strategic items among general items during normal export procedures.

As of 2020, North Korea's major trading partners are China, Russia, Vietnam, India, Nigeria, and Switzerland. Compared to 2019, Mozambique, Tanzania, Ghana, and Thailand entered the top 10, while Brazil, Bangladesh, Pakistan, and South Africa pushed out of the top 10. North Korea's trade dependence on China accounts for 88.2%, making it the largest trading partner for years, and it shows that North Korea is mainly conducting trade with Asian and African countries. North Korea's most important export items are mineral products (HS 25-27) and steel & metal products (HS 72-83) and the most significant import items are mineral products (HS 25-27) and oils & fats & prepared foods (HS 15-24). In 2017, due to UN Security Council sanctions for North Korea's international ballistic missile (ICBM) test-fire, North Korea's exports from 3 billion dollars fell by 90% to less than 300 million dollars. This is the result of most of North Korea's major export items included in the export ban, and changes have occurred in its export items. In 2020, export fell to less than 100 million dollars due to border lockdown measures to prevent the spread of COVID-19, which also affected the change of North Korea's major export items.

Although North Korea does not officially publish its foreign trade statistics, in order to review North Korea's trade information, KOTRA statistics are utilized. KOTRA statistics provide only two digits of HS code number, so it is challenging to identify detailed item classification. Moreover, these statistics are based on the export amount, so it is difficult to determine the exact quantity of export items. It is expected that information on North Korean trading partners and items will be used to predict potential transferable export methods of North Korea's strategic items when North Korea denuclearizes.

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Keywords: North Korea, Trading partners, Export items, Strategic items, KOTRA

A Study on Illegal Transfer Paths of North Korea's Strategic Items

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In the previous study, the types of North Korea's strategic items, foreign trading partners, and export items were investigated. From North Korea's typical trade paths, it is possible to predict the paths through which North Korea's strategic items are illegally exported upon denuclearization. Trading partners of North Korea are the potential importing countries or end-users of strategic items, which can be disguised or concealed as if it is general export items during typical export procedures. So, in this study, transfer paths of North Korea's export items are examined by utilizing KOTRA statistics, including item type HS code and its total price. Also, AnyLogic, a comprehensive simulation modeling tool, the simulation will be conducted to identify the paths for illegal transfer and calculate the time required.

The information on North Korea's trading partners and items is used for establishing export scenarios in which strategic items are transferred to other countries through North Korea's ports, airports, railroads, and roads. To be specific, China, Russia, and South Korea, countries that share a border with North Korea, export items transported only by land; the items will arrive first in the referred three countries. Since the types of items, North Korea transacts with each country are different, the total amount and frequency of transactions are different; the probability of strategic items being included in general export items and transferred during customs clearance also varies. Even if it does not border North Korea, North Korea can export items through ports to countries adjacent to the coastline, and North Korea can even export items to any country by airspace even if it is not adjacent to the coastline. So, all publicly open ports, airports, railways, and roads are surveyed. Their geographic information, such as EPSG 4326 and EPSG 3857 coordinate system, are applied to confirm and visualize valid export paths starting from North Korea.

In conclusion, effective export paths in North Korea are identified based on North Korea's each major transportation hub by using AnyLogic simulation. It is possible to predict the paths through which North Korea's strategic items will be transferred by combining information on major export items and countries that North Korea mainly transacts with.

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Keywords: North Korea, Strategic items, Illegal transfer paths, AnyLogic

Acquisition Path Analysis Method Derived Using Physical Model

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KINAC is trying to build a comprehensive aerial view of the nuclear material balance to predict North Korea's weapons-grade nuclear material production capacity. We are creating a visualization model for North Korea's nuclear facilities as part of these efforts. However, information on North Korea's nuclear facilities is scarce, and it is not easy to consider additional facilities other than those already known. In addition, in the case of a model that targets only exceptional situations, it is not easy to secure objectivity for model validation, so it is necessary to upgrade to a general-purpose analysis tool that can be applied more generally.

The following two examples are proposed as an analysis tool that can be a high degree of analysis.

The first case is an Acquisition Path Analysis (APA) utilized to introduce IAEA's State-Level Approach (SLA). The acquisition path analysis aims to find and evaluate the technically possible pathways to obtain nuclear materials for nuclear weapons or other nuclear explosive development. It can be an acquisition route if it is possible to produce at least 1 Significant Quantity (SQ) of weapon-grade nuclear material within five years. The assessment of technologically feasible pathways is based on available information about the country's past and present nuclear cycle capabilities.

The second is the IAEA Physical Model. The IAEA Physical Model was carried out to introduce a comprehensive approach to all information on a country's nuclear activities. It describes and characterizes the technologies and processes expressed at all levels of the acquisition path, depending on the development objectives. The IAEA Physical Model attempts a multi-tiered acquisition path analysis to identify all known technologies and processes in the nuclear fuel cycle, from raw material production to weapon usable material acquisition. Based on this analysis, the IAEA evaluates the signs of nuclear proliferation in a specific country.

Based on the two cases discussed above, we intend to derive the following implications and priorities for extending the existing nuclear cycle model to a more general-purpose for a specific country.

First of all, the requirements necessary to evaluate nuclear non-proliferation or verification of denuclearization must be at a level that the international community can recognize. In the stage of actual denuclearization verification, since verification will be conducted through the IAEA, a corresponding level of tools and technology will be required. From this point of view, the following is presented as a prerequisite for adding versatility to the existing physical model:

It is necessary to derive all processes related to the nuclear cycle and standardize relevant indicators and data. In order to determine the signs of nuclear activity, detailed information on technologies, materials, by-products, and wastes, which are essential for each process, is required. For denuclearization verification, cumulative information from the past to the time point is required, and a comparative analysis of the operation history information of all facilities and the amount of nuclear material is required. To this end, it is necessary to make it possible to trace the history at every point where it can be determined that nuclear material has been diverted so that missing nuclear material can be found. Based on this, it is expected that it can be possible to evaluate a hypothetical threat state, but it is also expected that it will be easy to verify the model through the evaluation of easily accessible domestic facilities.

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Keywords: Nuclear non-proliferation, IAEA physical model, Acquisition path analysis

A Study on the Establishment of an Integrated System for Nuclear Import and Export Control

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With the rapid improvement of hardware and software-related IT technology, applying A.I. to the private and public sectors, such as the Food Poisoning Prevention Program in Nevada and Smart City based on big data in Boston, is steadily increasing. However, the cases of application to the regulation sector of government are still insufficient. The Korea Institute of Nonproliferation and Control (KINAC) is studying to apply A.I. technology to the regulation to improve the objectivity, consistency, and efficiency of classification and export licensing review. The KINAC developed the Nation Nuclear Technology Information Collection and Analysis System using A.I. techniques such as machine learning and deep learning techniques. KINAC and FNC Technology are developing the Export Risk Assessment System using A.I. modules and Bayesian Networks. The KINAC and Korea Atomic Energy Research Institute (KAERI) are developing an inventory history management system subject to the Nuclear Cooperation Agreements. The Nuclear Safety and Security Commission (NSSC) and KINAC operate the Nuclear Import and Export Control System (NEPS) for application and export license review according to relevant laws such as the Foreign Trade Act. Therefore, preparing an integration plan for the existing NEPS and the new systems is necessary. Since the NEPS has to be operated and accessible at all times, so the stability of the NEPS is the most important when integration and linking. So, it is suggested that the Collection and Analysis System and the Risk Assessment System, which require a lot of data traffic, are configured in a server separate from the NEPS, and the new DB and the NEPS DB are only linked. An inventory history management system is also suggested to be integrated and configured into the NEPS. Third, it is recommended that each system lists the information provided to or received from the NEPS in advance, and one-way communication should be performed basically. Two-way communication should be performed when necessary. Finally, against various cyber accidents and information leakage, it is proposed to review security vulnerabilities and apply essential security measures and guidelines. Through the integration and linkage of these systems, it is expected that the objectivity, consistency, and efficiency of classification and export licensing review of the KINAC are strengthened, and national transparency of development, production, and use of nuclear material is enhanced. It can be satisfied with the increasingly strengthened demands of the international community on duty for strategic item management.

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Keywords: Nuclear export, Nuclear import, System integration, System configuration, Artificial intelligence, NEPS

Selection of Cyber Security Exercise Target System for Unauthorized Removal of Nuclear or Other Radioactive Material and Sabotage of Nuclear Facilities

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In accordance with the Enforcement Decree of the Act on Physical Protection and Radiological Emergency, operators of Nuclear Power Plants (NPP)s must conduct full cyber security exercise once a year and partial exercise at least once every half year. Nuclear operators need to conduct exercise on systems with high attack attractiveness in order to respond to the unauthorized removal of nuclear or other radioactive material and sabotage of nuclear facilities. Nuclear facilities identify digital assets that perform SSEP (Safety, Security, and Emergency Preparedness) functions as CDA (Critical Digital Assets), and nuclear operators select exercise target systems from the CDA list and perform the exercise. However, digital assets that have an indirect impact (providing access, support, and protection) from cyber attacks are also identified as CDAs, and these CDAs are relatively less attractive to attack. Therefore, guidelines are needed to select the exercise target system in the case of unauthorized removal of nuclear or other radioactive material and sabotage response exercise. In the case of unauthorized removal of nuclear or other radioactive material, these situations cannot occur with cyber attacks and external factors such as terrorists must be taken into consideration. Therefore, it is necessary to identify the list of CDAs that terrorists can use for cyber attacks among CDAs located in the path of stealing and transporting nuclear material and conduct intensive exercise on these CDAs. A typical example is a security system that can delay detection when terrorists attack facilities. In the case of sabotage exercise, a safety-related system that causes an initiating event by a cyber attack or failure to mitigate an accident in a DBA (Design Basis Accident) situation should be selected as an exercise target. It is difficult for sabotage to occur through a single cyber attack because a nuclear facility has several safety concepts such as redundancy, diversity. Therefore, it can be considered to select an exercise target system under the premise of not only a cyber attack but also a physical attack. In the case of NPPs, it is assumed that LOOP (Loss of Offsite Power) has occurred, and CDA relationships to accident mitigation can be selected as an exercise target. Through exercise on the CDA, which is more associated with unauthorized removal of nuclear or other radioactive material and sabotage of nuclear facilities, it is expected to review the continuity plan and check systematic response capabilities in emergencies caused by cyber attacks.

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Keywords: Cyber security, Cyber attack, Exercise, Unauthorized removal of radioactive material, Sabotage

Threat Information Collection and Analysis Method for Developing Cyber Security Threat Assessment Report

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KINAC (Korea Institute of Nuclear Non-proliferation and Control) is entrusted with the NSSC (Nuclear Safety And Security Commission) to conduct threat assessments for nuclear facilities. As part of the threat assessment, DBT (Design Basis Threat) must be established every three years, and a threat assessment report must be developed for DBT establishment. This paper suggests a method for collecting and analyzing cyber threat information for the development of a cyber security threat assessment report. Recently, cyber threats not only in the IT (Information Technology) field but also in the ICS (Industrial Control System) field are rapidly increasing. As cyber threats increase, threat information including related attack techniques is also increasing. Although KINAC is conducting a threat assessment on cyber security at nuclear facilities, it cannot collect and analyze all cyber threat information. Therefore, it is necessary to determine a reliable source of threat information for threat assessment, and establish a strategy for collecting and analyzing threat information for DBT establishment. The first method for collecting and analyzing threat information is to first collect threat information on industrial fields with high similarity to nuclear facilities. Most of the disclosed cyber threat information is in the IT field, and most of this information is not suitable for closed-network nuclear facilities. Therefore, it is necessary to first collect and analyze threat information on facilities that use networks similar to nuclear facilities such as energy and financial sector. The second method is to analyze the attack technique for the collected threat information. The biggest factor in DBT reset is whether there is a new threat and how much it has increased compared to the existing threat. Therefore, it is necessary to analyze which attack technique was used in the collected threat information, and as part of the analysis, a cyber attack analysis model such as a kill chain can be used. The last method is to collect and manage the disclosed vulnerability information. In order to manage vulnerabilities, it is necessary to analyze what assets are in the nuclear facility first. By matching the reported vulnerability with the CDA (Critical Digital Asset) in the facility, it is possible to analyze whether the CDA can be affected by a cyber attack. As cyber threats continue to increase, it is necessary to analyze threat cases of similar facilities, attack techniques using attack models, and vulnerability analysis through asset identification in order to develop a threat assessments report.

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Keywords: Threat assessment, Cyber security, Cyber attack, Design basis threat

A Study on Criteria of Selecting Database When Establishing an Automated Mechanism of Collecting Publically-known Vulnerabilities for Nuclear Facilities

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Nuclear operators must sort out their digital assets as Critical Digital Asset (CDA) and manage their vulnerabilities. Since vulnerabilities are continually found and can be abused anytime, and the number of digital assets in nuclear facilities is increasing, it is important to collect publically-known vulnerabilities in automated mechanism to reevaluate their risks. KINAC is now in progress of establishing an automated mechanism of collecting publically-known vulnerabilities for nuclear facilities. This paper will discuss about criteria of selecting database when establishing an automated mechanism of collecting publically-known vulnerabilities for nuclear facilities. When selecting sets of vulnerability database, the characteristic of sets of digital assets need to be managed, importance of each digital asset, and where and who will use the set of digital assets should be mainly considered. Most of safety-related CDAs are made and used in the United States, and safety-related CDAs are similar to Information and Communication Technology (ICT) facilities. Therefore, the main vulnerability database used in the United States should be included when collecting the database of vulnerabilities. Especially, US government actively provides vulnerabilities of digital assets, enacting vulnerability disclosure policy to make each organization report their own potential vulnerabilities. The main vulnerability database of the US is National Vulnerability Database (NVD) of NIST. It contains over 150,000 vulnerabilities on ICT and Industrial Control System (ICS). Nuclear Energy Institute (NEI) published “Cyber Security Vulnerability and Risk Management”, Addendum 5 to NEI 08-09, and informed that US-CERT, ICS-CERT, and NVD can be used as publically-known vulnerability database, and US National Regulatory Commission (NRC) endorsed the publication. In South Korea, KrCERT and National Cyber Threat Intelligence (NCTI) share publically-known vulnerabilities, however, the number of vulnerabilities are less than those of NVD, and most of the data are duplication of those of NVD. Moreover, certain portion of information are only opened to authorized organizations, so it is unable to access those databases. Therefore, considering the fact that most information of vulnerabilities of CDAs are included in NVD and ICS-CERT, vulnerability database should also contain information from NVD and ICS-CERT. Otherwise, the database must contain equivalent information compared to NVD and ICS-CERT. Furthermore, the methodology for collecting vulnerabilities of digital assets from other countries is also required to be studied in the future research.

Keywords: Vulnerability, Verification, Publically-known vulnerability, Nuclear facility

Systematic Change Detection With Spectral Similarity Measures of SID for Uranium Tailing Piles to Monitor Suspicious Mining Activities in the Pyongsan Uranium Mine

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With the enhancement of the spatial resolution of satellite imagery (1 m or less), the satellite image analysis has been considered as the indispensable means for remote sensing of nuclear proliferation activities in the restricted access areas such as North Korea. Notably, in the case of an open-pit uranium mine, e.g. the Pyongsan uranium mine, the mining activity can be presumed if detecting the location and extent uranium tailing piles near shafts within temporal images. Several studies have researched on the target detection for minerals of interest such as limestone and coal to evaluate the economic activities by utilizing similarity measures, e.g., a spectral angle mapper and a spectral information divergence (SID). Thus, this paper presented a systematic change detection methodology for monitoring the uranium mining activity in the Pyongsan uranium mine with a similarity measure of SID. The proposed methodology using the target detection results consists of the following five steps. The first step is to acquire stereo images of areas of interest for change detection. The second step is to preprocess the stereo images as following measures: (i) the QUick Atmospheric Correction and the image-to-image registration with ENVI and (ii) the Gram-Schmidt pansharpening. The third step is to extract spectral information for minerals of interest, i.e., uranium tailing piles, by sampling pixels within the reference image. It is based on the satellite analysis report for the Pyongsan uranium mine by CSIS, which specified the location of the uranium tailing piles. As the fourth step, the target detection for uranium tailing piles was performed through the similarity measure of SID between the extracted spectral information and the spectral reflectance of the image. In the fifth step, the change detection was processed using the multivariate alteration detection algorithm, which compares the target detection results by canonical correlation analysis. Furthermore, this paper evaluated the performance of the proposed methodology with the change detection accuracy assessment index, i.e., the area under a receiver operating characteristic curve. In conclusion, this paper suggests the systematic change detection methodology utilizing time series analysis of target detection for uranium tailing piles, which can save time and cost for humans to interpret large amounts of satellite information at the restricted access areas. As future works, the feasibility of the proposed methodology would be investigated by analyzing distribution of minerals of interest regarding nuclear proliferation at Yongbyon, which has the historical events of suspicious nuclear activities.

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Keywords: Nuclear nonproliferation, Uranium tailing piles, Target detection, Change detection, MAD, ROC

Review of Current Inspection Methods and Research Trends for Applications of Nuclear Safeguards Inspection of Spent Nuclear Fuel

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Dry storage cask facilities are considered for temporary storage of spent nuclear fuels before their final disposal. According to relevant domestic laws and regulations, the integrity and gross defects of the PWR spent fuel must be inspected before they are transferred to the dry cask from a wet storage pool of a nuclear power plant. To meet nuclear safeguards requirements for a spent fuel transportation, the KINAC has been working to develop a simple and convenient Non-destructive Testing (NDT) equipment to verify the integrity and gross defects of the spent fuel assembly.

This study was conducted in two processes. The first stage is to review the current NDT techniques conducted in the nuclear fuel manufacturing process. During the manufacturing process, the Ultrasonic testing (UT) and Eddy Current Testing (ECT) technique are used for detecting the cracks or foreign materials in a cladding of a fresh fuel. During an over-haul period after an end of one fuel cycle, the sipping test of the spent fuel is performed for detecting the failed fuel assemblies. If it is determined through the sipping test whether any fuel assembly contains a failed fuel rod, the failed fuel rod of lots of fuel rods in the assembly is found out using the UT instrument. The ECT is used for detecting the internal defects and oxide layer thickness of a fuel cladding. Because the UT and ECT are the well-known technique and has already been employing for the spent fuel inspection, we adopted the UT and ECT technique for development of a new instrument for nuclear safeguards verification.

The second stage is to design the UT and ECT equipment in consideration of nuclear safeguards activities in the spent fuel pool. For nuclear safeguards inspection, irradiated fuel or non-fuel items are distinguished. Thus, verification equipment newly designed using the UT and ECT should detect not only a failed rod, but also a false tube, or a false rod, or a different material from a cladding. New probe and signal processing methods are developed to achieve these goals. The design of UT and ECT probes are preferentially carried out according to technical requirements – the probe thickness including a damper material should be less than 1.0 mm - and the study on analyzing signal distortion caused from material difference will be conducted for development of the safeguards inspection equipment. Detailed results of our study will be discussed in this conference.

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Keywords: Nuclear fuel assemble, Spent nuclear fuel, Safeguards, Non-destructive test, Ultrasound test, Eddy Current test

Development of the Bulk Handling Facility Technical Tour Software for SSAC Online Course

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Since 2017, the Korea Institute of Nuclear Nonproliferation and Control (KINAC) has been implementing State System of Accounting for and Control of Nuclear Materials (SSAC) training courses for the nuclear Newcomer States. This IAEA SSAC course for Newcomer States aims to provide overall concepts and techniques, particularly on nuclear material accountancy and control systems, and address future challenges with regard to developing new nuclear power plants. Due to the restricted travels and limited in-person access to training and facilities from the COVID-19 pandemic, however, a new software was developed to substitute a technical tour on bulk handling facility (BHF) of the training course, and the course was favorably shifted to online in 2021. This newly built training software allows participants to follow each step of the technical process at a virtual bulk handling facility, and provides a video tour for such conditions where the software is found difficult to operate. Another feature of the development is a security function that prevents access of unauthorized users to the software. The achievement is expected to strengthen the SSAC of Newcomer States and ensure the practical implementation of safeguards from the initial stage of their novel nuclear power program through cooperation with IAEA. This contribution of the Republic of Korea (ROK) as one of the leading countries in the field of nuclear nonproliferation will further extend the partnership between IAEA and ROK and promote cooperation with the Newcomer States.

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Keywords: IAEA, SSAC, COVID-19, Online training course, Software, BHF

A Review of Bayesian Statistics and its Application Direction for Nuclear Non-Proliferation

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Bayesian statistics, which is an approach to analyzing data based on Bayes' theorem, is currently widely used in all fields. However, it has been applied very limitedly to studies related to nuclear non-proliferation. Therefore, this paper provides a knowledge base and directions for using various Bayesian techniques in nuclear non-proliferation. First, the concepts and advantages of the Bayesian approach are summarized and the basic solving methods of Bayesian inference are explained. The Bayesian approach enables more precise posterior estimation using the prior probability and the likelihood functions. To solve Bayes' theorem, it is necessary to use the conjugate prior distribution, which is analytically solvable, or to use a numerical approach with computing power. Next, for several Bayesian statistics methods, the purpose of use and the mathematical derivation process are described. Bayesian linear regression analysis aims for obtaining a function that outputs the closest value to data of variables and results. Factor analysis is mainly used to derive a smaller number of unobserved latent variables that can represent observed variables. The logit and probit model are non-linear regression models for when the outcome is binary. The hierarchical model is to analyze by introducing hyper-parameters in an integrated manner when there are several groups of similar data. The Bayesian approach of these methods is generally based on the numerical solution of the Bayesian inference of the multivariate normal distribution. Finally, the previous researches that each introduced method have been applied to nuclear non-proliferation are investigated, and research topics that can be applied in the future are suggested. Bayesian statistics have been mainly used for precise estimation of the amount, location, and radioactivity spectrum of nuclear materials using detectors. Using Bayesian approach, it will be possible to perform various analyzes. For example, the change of activeness of nuclear program can be estimated by Bayesian inferences on the frequency and scale of nuclear tests. And it can be tried predicting the production of plutonium according to the core configuration and burnup using the Bayesian linear regression. Also, by introducing the Bayesian approach to factor analysis or logit analysis of nuclear development motives or nuclear proliferation probability, it can be expected to improve precision. With the development of computer technology, the use of Bayesian statistics increases rapidly. Based on the theory and applied topics summarized in this paper, it is expected that Bayesian statistics will be more actively used for nuclear non-proliferation in the future.

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Keywords: Bayesian statistics, Nuclear non-proliferation, Analysis method, Statistics, Data analysis, Review

Introduction on VR Nuclear Nonproliferation Education Content on HANARO Rx

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Korea Institute of Nuclear Nonproliferation and Control (KINAC) remains dedicated to providing national and international training to train the workforce in the area of nuclear nonproliferation and security. INSA has also provided a number of nuclear nonproliferation courses for the public such as middle or high school students and teachers, senior government officials, NGOs in the field of nuclear energy, and so on.

The recent trend calls for education with high field applicability. Additionally, as interest in nuclear nonproliferation has recently increased, the demand of the public for education is expected to be increased. However, since it is difficult for the public to access nuclear facilities, it is not easy to understand regulatory activities at nuclear facilities.

Therefore, KINAC has developed Virtual Reality (VR) content to enhance the public's understanding of nuclear facilities and on-site inspection activities of KINAC. VR technology is expected to be a new means that can enable the public to access nuclear facilities in spite of some "VR dizziness" usually called "Human Factor".

This paper introduces the composition and function of KINAC's nuclear nonproliferation VR content for Hanaro Rx in nuclear nonproliferation courses for the public and seeks ways to optimize it based on a one-year operation experience.

Keywords: Safeguards education, Virtual reality, VR, Public, COVID-19

2분과

사용후핵연료 처분전관리(Oral)

Overview of US DOE and Sandia Spent Nuclear Fuel Storage and Transportation R&D

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Sandia National Laboratories is the lead laboratory for the United States Department of Energy for the research and development (R&D) efforts to support the technical basis for the long-term storage, subsequent transportation, and permanent disposal of commercial spent nuclear fuel and high-level waste. Sandia does not design nuclear facilities; Sandia performs R&D to help ensure facilities and the fuel cycle are safe, sustainable, and secure. This talk will focus on the spent fuel storage and transportation programs that contribute to this work.

The goal in spent fuel storage and transportation R&D is to understand the mechanical integrity of the fuel, cladding, and storage system beyond interim storage and into disposal time frames. Our research is focused on understanding the high burn-up cladding integrity over time, understanding the thermal behavior during drying and storage, understanding potential cladding oxidation pathways, and quantifying in the external loads experienced during transportation, handling, and seismic events. Additionally, this work includes extensive work to understand the basic science of canister stress corrosion cracking and the potential consequences of a through wall canister crack.

Acknowledgements

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Keywords: Spent nuclear fuel, Transportation and storage, Cladding integrity, Stress corrosion cracking

Fatigue Analysis of Spent Nuclear Fuel for Normal Condition of Ship Transportation

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The purpose of this study is to develop the analysis procedures for the evaluation of the structural integrity of the spent fuel in normal condition of transport at sea. Spent nuclear fuel must be transported from the wet storage facility in the nuclear power plant to the intermediate storage facility, and the structural integrity must be maintained in vibration and shock loads during the transportation. In general, the transport of spent nuclear fuel is performed in three kinds of modes: road, rail, and sea. During transport, the spent nuclear fuel is subjected to repeated vibration and shock loads by road surfaces, railroad tracks, and waves of the sea. It should be evaluated whether the structural integrity of the spent fuel is maintained under these load conditions. All nuclear power plants in Korea are located in coastal sites, and the interim storage facility for spent nuclear fuel is highly likely to be decided as a coastal site as well. Therefore, the main mode of the spent nuclear fuel transport is expected to be maritime transport by ships. In this study, the analysis procedure was developed to evaluate the safety of spent fuel at maritime transport by ships, and the procedure for evaluating the integrity of spent fuel under normal conditions of maritime transport were proposed. CFD analysis using SeaFEM was performed for the vibration analysis of the ship by waves, and the structural vibration analysis of the transport system was simulated using the developed in-house codes. The fatigue durability of the cladding was also evaluated using the developed fatigue analysis program and the fatigue analysis used the strain data obtained from the structural analysis. It was concluded that the value of the fatigue damage on the spent fuel cladding during normal conditions of maritime transportation is close to “0” and the structural integrity of the spent fuel is maintained in the same condition.

Acknowledgements

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Keywords: Maritime transportation, Spent nuclear fuel, Vibration and shock, Fatigue analysis

Preliminary Design of Thermal Test Equipment of Dry Canister for Equivalent Fuel Assembly

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The temperature of the spent fuel cladding is the basis for the evaluation of integrity. It is almost impossible to directly measure the temperature of spent nuclear fuel. Because spent nuclear fuel is dangerous. We are preparing a test to measure the cladding temperature with an equivalent fuel assembly by simulating the characteristics of spent nuclear fuel. PLUS7 was selected as the test target in consideration of the amount of generation, thermal water retention, residual moisture content, and manufacturability of domestic spent nuclear fuel. The nuclear fuel assembly is planned to be manufactured in two main ways. Except for the cladding tube that simulates decay heat, the structure will be manufactured by KEPCO Nuclear Fuel, and fuel rods and canisters will be manufactured by SUKEGAWA Electric in Japan. The same nuclear fuel assemblies as commercial skeleton will be applied. The temperature of the fuel cladding will be measured by attaching a thermocouple directly to the surface of the cladding tube. The canister is composed of a basket, a basket supporter, a heater and drain tube, a lead, and an observation window. The working fluid is water and helium, and the maximum pressure inside the canister is 1.1 MPa and the minimum pressure is 0.05 kPa. The maximum temperature of the surface of the cladding was designed to be 500°C, and the maximum temperature of the sealing to keep airtightness was designed to be 250°C. To satisfy this condition, we plan to evaluate the leak rate below 10^{-5} std.cm³·s⁻¹, which is equivalent to helium tightness. The maximum heat of decay per fuel rod is 13 W, and one assembly is up to 3 kW. Production of the test equipment is expected to be completed in the first half of next year, and testing is scheduled to begin in the second half of next year. The test will evaluate all environments that the spent nuclear fuel may experience, such as dry normal conditions, abnormal conditions, wet conditions, and dry conditions. All data will be used for interpretation verification purposes.

Acknowledgements

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Keywords: Thermal test, Fuel cladding temperature, Spent nuclear fuel, Equivalent fuel assembly, Dry storage, Vacuum drying

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

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Molten Salt Reactor (MSR) is one of the generation-IV advanced nuclear reactors in which high-temperature molten salt mixture is used as the primary coolant, or even the fuel itself unlike most nuclear reactors that adopt solid fuels. The MSR has received a great attention because of its excellent thermal efficiency, high power density, and structural simplicity. In particular, since the MSR uses molten salts with boiling points higher than the exit temperature of the reactor core, there is no severe accident such as a core melt-down which leads to a hydrogen explosion. In addition, it is possible to remove the residual heat through a completely passive way and when the fuel salt leaks to the outside, it solidifies at room-temperature without releasing radioactive fission products such as cesium, which make the MSR inherently safe. Both fluoride and chloride mixtures can be used as liquid fuel salts by adding actinide halides for MSRs. However, the MSRs using chloride-based salt fuels can be operated for a long time without adding nuclear fuel or online reprocessing because the actinide solubility in chloride salts is about six times higher than that in fluoride salts. Therefore, the chloride-based MSRs are more effective for the transmutation of long-lived radionuclides such as transuranic elements than the fluoride-based MSRs, which is beneficial to resolve the high radioactive spent nuclear fuel generated from light water reactors (LWRs). This paper examines liquid fuel fabrication using an improved U chlorination process for the chloride-based MSRs and presents the strategy for the management of gaseous fission products generated during the operation of MSR.

Acknowledgements

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Keywords: Molten salt reactor (MSR), Liquid fuel, Fission products, Chlorination, Spent nuclear Fuel, Pyroprocess

Mechanical Loads on Spent Nuclear Fuel in Storage and Transportation Scenarios

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This presentation summarizes recent research on estimating the mechanical loading environment of spent nuclear fuel (SNF) during normal storage and transportation scenarios sponsored by the US Department of Energy Spent Fuel and Waste Science and Technology (SFWST) program. Normal conditions of truck, ship, and railroad transportation of SNF were studied with testing and numerical modeling to determine that the shock and vibration loads applied to SNF during transportation are not expected to challenge SNF cladding integrity or the fatigue life of cladding. The 30 cm package drop scenario was studied with experiments and modeling to determine that mechanical loads during a 30 cm SNF package drop scenario are only expected to challenge SNF cladding integrity under worst-case conditions at elevated temperatures. The SFWST program is currently preparing seismic shake table testing to record SNF mechanical loads in a dry storage earthquake scenario. This presentation summarizes the findings of the transportation and package drop research and details the progress made on the current seismic test.

Keywords: Spent nuclear fuel, Transportation and storage, Cladding integrity, Mechanical load

The Preliminary Study of Probabilistic Safety Assessment for Intermediate Storage Facility

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In South Korea, the master plan for high-level radioactive waste management, announced in 2016, suggested the construction and operation of intermediate storage facilities on a permanent disposal site and specified the adoption of dry storage in consideration of the ease of operation and expansion. As of 2021, the government is again reviewing its overarching policy on the back-end fuel cycles, including intermediate storage and permanent disposal. In the case of dry storage facilities, safety evaluation is being conducted using a combination of deterministic and probabilistic approaches, similar to that of nuclear power plants. The two methods are complementary, of which Probabilistic Safety Assessment (PSA) has the advantage of being able to identify key scenarios affecting safety, but its use in storage facilities has not been highlighted so far. However, depending on the spent fuel management phases such as loading, transportation, and storage, it may be not enough to capture effective and efficient safety evaluation only deterministically, and probabilistic methods may contribute to the evaluation of long-term operation or external events such as an earthquake.

There have already been cases where PSA has been performed on a part of the nuclear fuel cycle through previous studies. This paper created the safety assessment model based on open sources such as the released EPRI reports, by targeting arbitrary intermediate storage facilities. The model considered the scenarios for loading, transportation, and storage, with human error respectively. It will be able to be modified and improved to fit domestic and specific intermediate storage facilities in the future.

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Keywords: Probability safety assessment, Intermediate storage facility, Loading, Transportation, Storage

Development of Failure Prediction Model for Zircaloy-4 Cladding With Hydride Under Pinch Load

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The evaluation of the damage ratio of spent nuclear fuel is a very important intermediate variable for dry storage risk assessment, which requires an interdisciplinary and comprehensive investigation. It is known that the pinch load applied to the cladding can lead to Mode-3 failure and the cladding becomes more vulnerable to this failure mode with the existence of radial hydrides and other forms of mechanical defects. In this study, the failure resistance of Zircaloy-4 cladding against the pinch load is investigated using numerical simulations assuming the existence of radial hydrides. The simulation model is based on the microscopic images of cladding. A pixel-based finite element model was created by separating the Zircaloy-4 and hydride using the image segmentation method. The image segmentation method uses a morphology operation basis, which is a preprocessing method through erosion operation after image expansion to enable normal segmentation by emphasizing pixels corresponding to hydrides. The segmented images are converted into a finite element model by assigning node and element numbers together with corresponding material properties. Using the generated hydride cladding finite element model, several numerical methods are investigated to simulate crack propagation and cladding failure under pinch load. Using extended finite element (XFEM) models the initiation and propagation of a discrete crack along an arbitrary, solution-dependent path can be simulated without the requirement of remeshing. The applicability of fracture mechanical parameters such as stress intensity, J-integral was also investigated.

Keywords: Spent nuclear fuel, Cladding, Hydride, Pinch load, Mechanical integrity

Thermal Analysis of Cylindrical Model of Waste Form Containing High Heat-generating Nuclides Using Analytical and Numerical Methods

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The fabrication of waste forms with high thermal and structural stability is an essential technology for the safe management and disposal of radioactive wastes. In particular, the thermal characteristics of waste forms containing high heat-generating nuclides such as Cs and Sr can be used for the optimized design of the waste form to secure its thermal safety, and they also provide basic design data for the safe management of canisters, storage systems, and repositories. The Korea Atomic Energy Research Institute is actively developing processes and equipment for fabricating various types of high-level wastes into a stable glass or ceramic waste form. In previous research related to the thermal analysis of the waste form, a relatively simple analysis was performed by using the analytic solution of the one-dimensional steady-state heat conduction equation considering the decay heat properties of the waste. As a specific application study, the optimized diameter of the cylindrical glass waste form was proposed by evaluating the centerline temperature of the waste form. In this study, we extended previous research by introducing a more complicated model, and the main results are summarized as follows. First, an analytical solution was derived by applying the temperature-dependent thermal conductivity expressed in the general form of polynomial function to the one-dimensional heat conduction problem previously studied. Second, the two-dimensional axisymmetric steady-state heat conduction problem with a more realistic cylinder model with finite length was modeled and solved by using the finite element method via Matlab's PDE (partial differential equation) toolbox. Third, thermal analysis was performed on the SrTiO_3 waste form, selected as a stable form of strontium nuclide, using the developed analytical and numerical methods. The differences in the temperature distribution and computation time were evaluated through a comparative study of both solutions. Although the problem considered in this study could easily be solved by using commercial CFD software such as ANSYS or SolidWorks, a code-based program was developed to facilitate parametric design study in conjunction with optimization algorithms. The analysis results could be used to evaluate the thermal stability of waste form and to optimize the shape and size of the waste form in consideration of the design constraints of storage systems or repositories.

Acknowledgements

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Keywords: Thermal analysis, Cylindrical waste form, Heat-generating nuclide, Temperature-dependent thermal conductivity

2분과

사용후핵연료 처분전관리(Poster)

Recovery of Strontium From Salt Solution After Chlorination

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Strontium-90 is a high heat-generating nuclide in spent nuclear fuel. The removal of the nuclide separation is indispensable to reduce the burden of storage and disposal of high-level radioactive waste. Korea Atomic Energy Research Institute has developed the molten salt immersion technique to separate the strontium by the chlorination of the strontium oxide in molten salt. It is needed to separate the salt for the recovery of strontium from the salt solution after the chlorination reaction. In this study, it was investigated on the recovery of the strontium from the salt. Vacuum distillation was used for the separation of strontium from the molten salt. The vapor pressures of the candidate salts were calculated by HSC chemistry and the apparent evaporation rates (AER) were measured at 830°C to evaluate the salts for strontium recovery. The candidate salts were LiCl, KCl, MgCl₂, NaCl and CaCl₂. The AERs of MgCl₂ and NaCl were 1.9 and 1.3 g/cm²-h, respectively. Those two salts can be separated from the strontium compound even though the AER values are much lower than those of LiCl-KCl (~ 8 g/cm²-h). CaCl₂ salt was rarely evaporated (AER < 0.03 g/cm²-h) and it is not suitable to use as a strontium recovery salt. Therefore, MgCl₂, NaCl, LiCl and KCl can be regarded as candidates for a strontium recovery salt.

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Keywords: Strontium, Chlorination, Recovery, Molten salt

Origin of Fe Oxide Deposition on Zirconium Oxide With UV Irradiation

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The origin of Fe oxide deposition on zirconium oxide with UV irradiation has been investigated in this study. After 7 day corrosion in the flowing autoclave, Fe based oxide is formed on the zirconium oxide with UV irradiation at 260°C, 6 MPa DI water. Zircaloy-4 coupon is irradiated with a 200 mW·cm⁻² UV, and the dissolved oxygen level is maintained below 100 ppb, and dissolved hydrogen concentration is maintained as 2.5 ppm. Zircaloy-4 coupon supplied from Westinghouse is used for this study. MULTEQ version 4.0 developed by EPRI is adopted to simulate how ions dissolved in water can generate deposits on the zirconium oxide with UV irradiation. ICP-OES data after 30 d corrosion in the flowing loop experiment is used for input file for MULTEQ simulation. The system temperature is set as 260°C, and 2,592 L of water is considered the total amount water into the autoclave (0.06 mL·min⁻¹, 30 d). Total numbers of simulation run is set as 8, and the system pH at 260°C is 6.06. Oxidation potential after run #8 is -0.44 V. From MULTEQ simulation, most Fe is existed as Fe(OH)₃ and Fe(OH)₂, and Fe ions can also exist, but no Fe metal observed. 5.09×10^{-6} ppm (9.73 ppb) of Fe²⁺, 2.81×10^{-6} ppm FeOH⁺, and 3.77×10^{-9} ppm Fe(OH)₃⁻ are in the system. It can be concluded Fe is existed as ion or hydroxide form in the solution. Two precipitates are found from MULTEQ simulation, First, NiO(s) = 5.21×10^{-5} g (52.1 µg), NiFe₂O₄ = 8.06×10^{-5} g (80.6 µg), and still they are negligible amount. The total concentration of Fe in the electrolyte is the summation of each Fe species concentration and it is equal to 2.69×10^{-4} ppm. This value is equivalent to 0.269 µg·kg⁻¹ in the solution. The total water volume of the 30 d experiment is 2,592 L (considering water flow from high-pressure pump), so the amount of Fe from ICP-OES data and MULTEQ results in 2,592 L electrolyte is 697.2 µg. This value is order of magnitudes higher than the mass of Fe from the deposits, which was already an upper estimate based on the assumptions. This clearly shows that Fe ions dissolved in the electrolyte can be the source of Fe₃O₄ on Zr oxide during corrosion with UV irradiation.

Keywords: UV Irradiation, Zirconium oxide, High temperature corrosion, MULTEQ Simulation

Simulated Damaged Nuclear Fuel Rod Manufacturing Device

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In general, if a nuclear fuel cladding tube is damaged during reactor operation, it is called fuel failure. If the cladding tube is damaged, the function of sealing the nuclear fuel material is lost, and the fission products accumulated inside the nuclear fuel rod may leak into the coolant. The causes are the most damage caused by foreign substances in a coolant such as small iron wires, and GTRF (Grid-to-Rod Wear) due to a grid, end-plug welding defect, PCMI (pellet cladding mechanical interaction), and oxidation corrosion damage. In this study, a device of simulating friction damage and debris induced damage between grid-fuel rods, which are the main causes of cladding tube damage, was developed. An air vibrator was installed as a function to induce vibration of the nuclear fuel rod. Sandpaper was installed between the grid and the fuel rod to induce friction between the grid-fuel rods. Saw teeth were installed on the grid to induce damage to foreign substances. It is believed that the simulated damaged nuclear fuel rod can be manufactured through on-study to provide the simulated damaged nuclear fuel rod necessary for the stabilization study of the damaged nuclear fuel rod.

Acknowledgements

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Keywords: Simulated damaged fuel rod, GTRF, Debris induced fuel failure, Vibration, Sand paper, Saw teeth

Standard Reference Data for Oxidation of HANA-6 Cladding Material

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NFDC (Nuclear Fuel and materials Data Center) developed standard reference data for oxidation of HANA-6 cladding material. Thermo-gravimetric analyzer (TGA) was used to measure oxidation, and the measuring device was self-calibrated using standard materials. The oxidation amount of the HANA6 cladding was measured in an oxidizing atmosphere in the temperature range of 400 to 700°C. Through this, oxidation data, oxidation rate model equation, and graph were developed. The uncertainty factors were analyzed from the oxidation model. The expanded uncertainty of oxidation data was calculated by evaluating the uncertainty for each uncertainty factor. The oxidation data produced in this study was self-rated through deliberation by a specialized committee of NFDC and third experts. It was finally registered as a reference standard through the technical committee of the National Reference Standards Center. It is believed that the standard reference data developed in this study will be helpful for increasing reliability and stability evaluation of nuclear fuel and spent fuel.

Acknowledgements

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Keywords: Standard reference data, HANA-6, Oxidation, Uncertainty, Nuclear fuel and materials data center

Uncertainty Evaluation of the Thermo-Gravimetric Analyzer

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National Standard Reference Center from Ministry of Trade Industry and Energy at Dec.30 2008. The fields of designation were nuclear fuel and energy materials. NFDC produces standard reference data of nuclear fuel and materials. To ensure reliability of experimental data uncertainty should be estimated. There are two kinds of uncertainty: A-type uncertainty from tester and B-type uncertainty from experimental equipments. To reduce the former, the measurement should be repeated for sufficient amount of times, and to reduce the latter type uncertainty all equipment have to be calibrated. In this study the uncertainty evaluation process of thermo-gravimetric analyzer (TGA) data was developed. The self calibration was performed using the standard mass and correction factor was obtained. The measurement model of oxidation was established, factors affected to uncertainty was analyzed, uncertainty of each factor using sensitivity coefficient was evaluated, combined uncertainty was calculated, and expanded uncertainty using coverage factor was calculated. It is believed that the uncertainty evaluation process of TGA data developed in this study will be helpful for increasing reliability and stability evaluation of nuclear fuel and spent fuel.

Acknowledgements

This research was supported by the Technology Innovation Program (20016225, Development and Dissemination on National Standard Reference Data) funded by the Ministry of Trade, Industry & Energy (MOTIE, Korea).

Keywords: Standard reference data, HANA-6, Oxidation, Uncertainty, Nuclear fuel and materials data center

Thermal Test for Design Approval of the KTC-360 Transport Cask

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Currently, the HI-STAR 63 transport cask, developed to transport CANDU spent nuclear fuel from the wet storage pool to the dry storage facility which is called the MACSTOR/KN-400, has a transport capacity of 120 bundles, which is unfavorable when considering transportation costs and other related aspects. According to the 'Basic Plan for High-Level Radioactive Waste Management (draft)', the total amount of CANDU spent nuclear fuel is expected to be approximately 660,000 bundles. To safely and efficiently transport this amount to interim storage facilities, it is essential to develop a large-capacity transport cask. Therefore, we have been developing a large-capacity PHWR spent nuclear fuel transport cask, called the KTC-360 transport cask. According to the transport-cask related regulations, the KTC-360 transport cask was classified as a Type B package, and such packages need to maintain integrity under the normal transport and accident conditions described in these regulations. To prove the thermal integrity of this cask under the normal transport and accident conditions, high-temperature and fire tests were performed using a one-third slice model of an actual KTC-360 cask. The results revealed that the surface temperature of the cask was 62°C, indicating that such casks need to be transported exclusively. The highest temperature of the CANDU spent nuclear fuel was predicted to be lower than the melting temperature of Zircaloy-4, which was the sheath material used. Therefore, if normal operating conditions are applied, the thermal integrity of a KTC-360 cask could be maintained under normal transport conditions. The fire test revealed that the maximum temperatures of the structural materials, stainless steel, and carbon steel, were 446°C lower than the permitted maximum temperatures, proving the thermal integrity of the cask under fire-accident conditions.

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Keywords: Pressurized heavy water reactor, High-temperature test, Fire test, Thermal integrity, Transport cask

Predictive Model Development of Coastal Environments for Estimating CISCC in Spent Nuclear Fuel Storage System Using LSTM Algorithm

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The amount of temporarily stored spent nuclear fuel in South Korea will be reaching saturation in a near future. Therefore, it is an urgent issue to construct a spent nuclear fuel storage system. In order to construct the storage system, some coastal environmental characteristics such as temperature, pH, and chemical composition of sea water in South Korea have to be evaluated and predicted because they can affect in deterioration of the storage system. However, in South Korea, the coastal environmental characteristics of area where the storage system is likely to be built are not well established until now. In this study, a time-series deep-learning algorithm is developed using the Long-Short Term Memory (LSTM) algorithm to predict and evaluate the coastal environmental characteristics based on the well-established data from Korea Meteorological Administration (KMA) and Ministry of Oceans and Fisheries (MOF). As a result, by developing the predictive model to evaluate the coastal environmental characteristics, we intend to apply it for site evaluation to construct the spent nuclear fuel storage system or many other applications related to the nuclear as well.

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Keywords: Time-series data, Deep learning, LSTM, CISCC

~~Suggestion of Repair of Defective Fuel~~

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Canceled Poster

Design & Build of Chloride-Induced Stress Corrosion Cracking Simulation Equipment on the Welding Part of Canister

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

In this study, the current status of CISCC simulation tests performed around the world to build a test equipment for the CISCC simulation accelerated test is investigated, and based on this, the test conditions suitable for the simulation test and specimen specifications are selected to establish the test equipment. The settings were performed. In consideration of the set device requirements, the essential limiting conditions for device manufacturing were derived, and detailed design was performed to satisfy them, and it was used to build a CISCC simulation test device for welding parts.

The CISCC simulation test equipment requires performance to maintain the test temperature range of room temperature to 80°C and humidity 35 to 95%. In addition, it should be manufactured in consideration of humidity and temperature maintenance in the chamber of the complex corrosion tester, measures to prevent leakage of the connection part between the chamber and the salt water tank of the complex corrosion tester, and measures to supply stable salt water and maintain temperature in the salt water tank. Based on these contents, detailed specifications and design contents of the chloride stress corrosion cracking simulation test apparatus were presented in this study.

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

Preliminary Data Analysis of 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 were conducted from September 30 to October 4, 2021. In order to investigate amplification or attenuation characteristics, according to the load transfer path, a number of accelerometers were 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, strain gages were attached on surrogate fuel assemblies. A ship named “JW STELLA” which has similar deadweight (5,000 ton) of existing spent nuclear fuel transportation ships was used for the sea transportation tests. The ship is propelled by 1,825 hp two main engines with two 4-bladed propellers.

There are two major vibration sources in the ship. One is the vibration from waves and the other is the vibration from the engine and propeller system. The sensor locations on the ship were determined considering the vibration sources. The sea transportation test was performed for 5 days, the test data were measured successfully. The ship with the test model was departed from Changwon and sailed to Uljin, sailed west to Yeonggwang and then returned to Changwon. In addition to sailing on a designated test route, circulation test, braking/acceleration test, depth of water test, and rolling test were conducted.

As a result of the preliminary data analysis of the sea test, power spectral densities and shock response spectrums were obtained according to the different test conditions. The vibratory loads caused by the wave mainly occurred in the frequency range of 0.1 to 0.3 Hz. The vibratory loads caused by the propeller occurred near the n/rev rotating frequencies, such as 5, 10, 20 Hz etc. However, those frequencies are far from the natural frequencies of local mode of the fuel rods, so it is considered that the vibratory loads from the wave and the propeller do not have a significant influence on the structural integrity of the fuel rods. Among all the test cases, maximum strain occurred at SG31 near the bottom nozzle on the test; the magnitude was 73.62 micro strain.

Based on the analyzed road and sea transportation test data, a few input spectra for the shaker table test will be obtained and the shaker table test will be conducted in 2022. It is expected that the detailed vibration characteristics of the assembly which were difficult to identify from the test results can be investigated.

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

Survey of Dynamic Material Properties of Degraded Concrete in Dry Cask Storage System

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Concrete structures of spent nuclear fuel interim storage facility should maintain their shielding ability and structural integrity during normal, off-normal and accident conditions. The concrete structures may deteriorate if the interim storage facility operates for more than several decades. Even if deterioration occurs, the concrete structures must maintain 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.

In accident conditions such as tip over and aircraft collision, both static material properties and dynamic properties of the concrete are required to evaluate the structural integrity of the concrete structures. Unlike the calculated damage results for the static deformation of the concrete structure, it is very difficult to accurately estimate the damage values of the degraded concrete structures where an aircraft collides at a high strain rate. Therefore, the present authors have a plan to establish a database of the dynamic material properties of deteriorated concrete and implement to a Finite Element Analysis model.

Prior to that, dynamic increase factors described in a few technical specifications were investigated. The dynamic increase factor represents the ratio of the dynamic to static strength and is normally reported as function of strain rate. In ACI-349, only the strain rate is used as a variable in the empirical formula obtained from the test results of specified concrete strengths of 28 to 42 MPa. The maximum value of dynamic increase factor is limited to 1.25 in the axial direction and 1.10 in the shear direction. On the other hand, in the case of the CEB model, static strength is included as variables in addition to the strain rate, and a constitutive equation in which the slope changes from the strain rate of 30 /s is proposed. As plotting the two dynamic increase factor models, in the case of ACI, it is drawn as a single line, but in the case of CEB, it is plotted as multiple lines depending on the static strength.

The test methods and specimen sizes of the previously performed tests, which measured the concrete dynamic properties, were also investigated. When the strain rate is less than 10 /s, hydraulic or drop hammer machines were generally used and the length of the specimens was more than twice the diameter in most cases. However, in the case of Split Hopkinson Pressure Bar tests, the small size specimens are preferred to minimize the inertia effect, so the specimens were small and the length was less than twice the diameter.

We will construct the dynamic properties DB with our planned deteriorate concrete specimen test, and also include the dynamic property data already built in the previous studies.

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Keywords: Dry cask storage system, Interim storage facility, Concrete deterioration, High strain rate

Reaction of Rare Earth Oxide From the Simfuel and Chlorination Reagents in LiCl-KCl Molten salt

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To estimate the removal efficiency of TRU and rare earth elements in an oxide spent fuel, basic dissolution experiments were performed for the reaction of rare earth elements from the prepared simfuel with chlorination reagents in LiCl-KCl molten salt. Based on the literature survey, NH_4Cl , UCl_3 , and ZrCl_4 were selected as chlorination reagent. CeO_2 and Gd_2O_3 powders were mixed with uranium oxide as a representative material of rare earth elements. Simfuel pellets were prepared through molding and sintering processes, and mechanically pulverized to a powder form. The experiments for the reaction of the simfuel powder and chlorination reagents were carried out in a LiCl-KCl molten salt at 500°C . To observe the dissolution behavior of rare earth elements, molten salt samples were collected before and after the reactions, and concentration analysis was performed using ICP. After the reaction completed, the remaining oxide was washed with water and separated from the molten salt, and XRD was used for structural analysis. As a result of salt concentration analysis, the dissolution performance of rare earth elements was confirmed in the reaction experiments of all chlorination reagents. In an experiment using NH_4Cl and ZrCl_4 , the uranium concentration in the molten salt was also measured. In other words, it seemed that not only rare elements but also uranium oxide, which is a main component of simfuel, was dissolved. Therefore, it is thought that the dissolution of rare earth elements is also possible due to the collapse of the uranium oxide structure of the solid powder and the reaction with the oxide of rare earth elements exposed to molten salt. As a result of analyzing the concentration changes of Simfuel before and after each reaction, there was little loss of uranium and rare earth elements (Ce/Gd) in the NH_4Cl experiment, but a significant amount of rare earth elements were found to be reduced in the UCl_3 experiment, and a large amount of rare earth elements were reduced in the ZrCl_4 reaction.

Keywords: Uranium oxide, Rare earth oxide, Dissolution, Spent nuclear fuel, Molten salt

Experimental Technology of Long-term Cooling Effect for Hydride Reorientation Test of Non-irradiated Zircaloy-4 Cladding

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A long-term cooling effect on hydride reorientation of a cladding tube can affect the integrity of spent nuclear fuel transportation and long-term storage. In this study, experimental setup for investigating the degree of radial reorientation of hydrides in the circumferential direction during the long-term cooling was established. The experimental setup was designed to be simplified since the long-term evaluation requires a long term period such as 12, 18 and 24 months when the cladding tube specimen is gradually cooled down from 400°C to 100°C. For the test, hydrogen-charged specimens of 100 ppm, 200 ppm, and 500 ppm were prepared. The specimen was sealed with fixtures and check valve, and was pressurized up to 90 Mpa. To heat the specimen, a box-type furnace was used while the temperature of the specimen was measured from thermocouples attached to the specimen. After the heat treatment, the long-term cooling was performed by developing temperature control program to investigate several cooling rate conditions of the specimen. As a reference case, microstructure and brittle property of the hydrogen-charged specimens of 100 ppm, 200 ppm, and 500 ppm without the long-term cooling was observed. In the case of the hydrogen content, it was uniformly distributed in circumferential direction although it was non-uniform in the axial direction. In the case of the brittle property, a compression test was performed. For the future work, the microstructure and brittle property of the hydrogen-charged specimens after the several long-cooling conditions were investigated. Then, the degree of radial reorientation of hydrides in the circumferential direction during the long-term cooling was studied.

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Keywords: Hydride reorientation, Coolingspeed, Ring compression test, Long-term cooling

**~~Application of Independent Air-cooled System Installation on Spent Fuel Pool
for Decommissioning of Pressurized Water Reactor Kori Units 3 and 4~~**

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Canceled Poster

Development of Rod Performance Evaluation Code to Resolve Integrity Issues of CANDU Spent Fuel

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Prior to the investigations on fuel degradation it is necessary to describe the reference characteristics of the spent fuel. It establishes the initial condition of the reference fuel bundle at the start of dry storage. In a few technology areas, CANDU fuels have not yet developed comprehensive analysis tools anywhere near the levels in the LWR industry. This requires significantly improved computer codes for CANDU fuel design. In KNF, in-house fuel performance code was developed to predict the overall behavior of a fuel rod under normal operating conditions. It includes the analysis modules to predict temperature, pellet cracking and deformation, clad stress and strain at the mid-plane of the pellet and pellet-pellet interfaces, fission gas release and internal gas pressure. The main focus of the code is to provide information on initial conditions prior to dry storage, such as fission gas inventory and its distribution within the fuel pellet, initial volumes of storage spaces and their locations, radial profile of heat generation within the pellet, etc. Potential degradation mechanisms that may affect sheath integrity of CANDU spent fuel during dry storage are: creep rupture under internal gas pressure, sheath oxidation in air environment, stress corrosion cracking, delayed hydride cracking, and sheath splitting due to UO_2 oxidation for a defective fuel. To upgrade the developed code that address all the damage mechanisms, the first step was a review of the available technical information on phenomena relevant to fuel integrity. The second step was an examination of the technical bases of all modules of the in-house code, identify and extend the ranges of all modules to required operating ranges. Further improvements being considered include upgrades of the analysis module to achieve sufficient accuracy in key output parameters. The emphasis in the near future will be on validation of the in-house code according to a rigorous and formal methodology. The developed models provide a platform for research and industrial applications, including the design of fuel behavior experiments and prediction of safe operating margins for CANDU spent fuel.

Keywords: CANDU spent fuel, Fuel performance code, Potential degradation mechanism, Dry storage

Stress-based Mitigation of Chloride-Induced Stress Corrosion Cracking for Spent Fuel Dry Storage Canister

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The spent fuel dry 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 canister on the dry storage system has several welding lines in the wall and lid, which have high residual tensile stresses after welding procedure. Interaction between stainless steel and chloride environment from a sea results in an aged-related degradation phenomenon causing chloride induced stress corrosion cracking (CISCC) in the dry storage system. A pending issue to the interim storage of spent fuel awaiting repository disposal is their susceptibility to CISCC of stainless steel canisters. The available mitigation technology should be studied sufficiently to prevent the degradation phenomenon. This paper assesses stress-based mitigation to control residual tensile stress practically applicable to the atmospheric CISCC for the aging management of the stainless steel canisters.

There are major components, that is, elevated tensile stress, susceptible material and corrosive environment that must be simultaneously present for CISCC degradation to occur. Surface stress improvement can effectively mitigate the potential for CISCC of the canister external surfaces. The potential deleterious effect of the additional work is negated by the presence of compressive residual stress, which removes the tensile stress needed for CISCC to occur. Surface stress improvement methods such as shock-based peening, shot peening and low plasticity burnishing can be applied for surface stress improvement of the canisters. Stress relaxation processes and advanced welding methods such as laser beam welding and friction stir welding can be also available to mitigate the susceptibility to CISCC. As the result assessing the stress-based mitigation technologies, promising candidate methods could be selected to reduce the residual tensile stresses and to control an aged-related degradation condition causing CISCC in the spent fuel dry storage canister.

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Keywords: Spent fuel dry storage canister, Austenitic stainless steel, Chloride induced stress corrosion cracking, Stress-based mitigation, Residual tensile stress

Comparison Between Thermal Analysis Results With FVM and Sub-channel Codes in a Single Assembly Heat Transfer Test

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Dry storage is a predominantly used method as a spent nuclear fuel storage system after spent nuclear fuel is cooled in the spent fuel pool. Spent nuclear fuel is highly radioactive and it generates heat called decay heat originated by fission products and radiation. Therefore, temperature of spent nuclear fuel should be predicted whether its cladding temperature is maintained under 400°C, which is the allowable temperature limit of cladding in a dry storage.

ANSYS Fluent and COBRA-SFS are predominantly used computational method to investigate the temperature of spent nuclear fuels in a dry storage. Herein, thermal analysis results with the methods were compared based on a Single Assembly Heat Transfer Test, which is a heat test with an electrically heated model of a single PWR fuel assembly in a dry cask performed at the Pacific Northwest Laboratory. Decay heat was 1kW and backfill gas was air. Fix temperature boundary condition is applied to inner shell according to measured temperature.

In case of peak cladding temperature (PCT), Fluent predicted 240–284°C, while COBRA-SFS gave 243–292°C. The discrepancy between the codes is under 2.5%. The location where PCT took place was 3.65 m from the bottom of the assembly in both results. However, temperature difference between the upper and lower region of the assembly based on the Fluent was smaller than the temperature difference based on the COBRA-SFS. It means the heat was well transferred in an axial direction with Fluent compared to COBRA-SFS. In lower plenum region where air was naturally circulated, COBRA-SFS had disadvantages compared to Fluent because it modeled the lower plenum by single node, so it was hard to simulate convection heat transfer by natural circulation. natural circulation speed of air in a center region of the assembly was 0.07–0.1 m·s⁻¹ in both cases.

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Keywords: Spent nuclear fuel, Dry storage cask, Thermal analysis, CFD simulation, Sub-channel code

Selection of Safety Information for the Delivery of PWR Spent Nuclear Fuel

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The manufactured nuclear fuel assembly is loaded into the nuclear reactor after the core design, and is finally discharged to the wet storage pool after depletion for 3 cycles. The discharged spent nuclear fuel is transported and stored in a dry storage system at the on-site of the nuclear power plant, which is cooled by natural convection, and undergoes final disposal or reprocessing through an intermediate dry storage facility. In this series of processes, the characteristics of the final product, the spent fuel, vary depending on the environmental conditions, so it is essential to manage each history data to verify the long-term integrity of the spent nuclear fuel. In this paper, safety information on spent nuclear fuel is described in order to establish technical requirements that should be considered in each stage of storage, transport, reprocessing, and disposal of spent nuclear fuel. Comprehensive safety information on spent nuclear fuel is basically calculated from basic information that considers characteristic information that can be obtained through the manufacture and design of nuclear fuel assemblies, operation history in a nuclear reactor, and location history in a wet storage pool. It can be divided into secondary production information (SF Burnup, Nuclide Inventory, etc.) and tertiary integrity-related information obtained through cladding inspection during spent fuel storage. KHNP produces this multi-layered information according to the production stage and manages it through the comprehensive management system of the spent nuclear fuel, and safety information with some errors is not only improved through re-verification but also continuously updated. In this paper, the spent nuclear fuel safety information was derived based on various information calculated in the entire process of being discharged and managed in a wet storage pool, including new fuel manufacturing information and depletion history. Such safety information will be used as basic data for long-term safe management of spent nuclear fuel, and will be continuously produced and managed. In the future, additional discussions will be held on the safety information of the spent nuclear fuel through consultation with KORAD and regulatory agencies.

Keywords: Spent fuel, Dry storage, PWR, Fuel assembly, Burnup, Safety information

Design Requirements for Fuel Storage and Handling Area at Nuclear Power Plant

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The design of nuclear fuel storage and handling area includes the activities related to the storage and inspection before fuel loading, transfer into the reactor, removal of irradiated fuel to the spent fuel storage rack, underwater handling and storage, and handling into a shipping cask. The purpose of this study is to provide the design requirements for the spent fuel pool to be prevented from the loss of cooling water and for heavy load control to prevent any load drop resulting in damage to safety-related systems during heavy load handling in accordance with the regulatory guidelines. And another purpose is to review the sizing of minimum wet storage capacity in the spent fuel pool based on the maximum refueling batch from the core during refueling plus a full core off-load of fuel assemblies and the minimum discharge burnup spent fuel storage during the design life of plant requested by the utility. As the results of this study, the current general arrangement for the spent fuel storage and handling area and the minimum storage capacity are evaluated. These can be good recommendations to enhance more safe and efficient if implemented to the new nuclear power plants.

Keywords: Fuel storage and handling area, Fuel handling, Spent fuel pool

Chlorination of Simulated Oxide Fuel to Minimize Volume of High Level Nuclear Fuel Waste

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Facing the problem of saturation of spent nuclear fuel (SNF) stored in temporary storage facilities on sites, interest in the treatment of SNF is increasing, and it is recognized as a task that needs to be solved promptly. Although direct disposal is a general method for dealing with SNF, the entire fuel assembly is classified as high-level waste; thus, the burden of disposal is high. In order to minimize the disposal burden with enhancing safety for long term storage, it is necessary to develop SNF treatment technologies and continuous efforts are required from a national policy perspective. The present study focused on minimizing the volume of high level waste from light water reactor fuel by separation of uranium, which accounts for most of SNF. The chlorination characteristics of uranium (U), rare earth (RE) oxides were confirmed through lab-scale experiments, and the possibility of uranium separation from U-RE simulated fuel was evaluated using NH_4Cl chlorinating agent. The detailed results will be posted and discussed.

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

Molten Salt Immersion Technique With MgCl_2 Based Molten Salts for the Sr Nuclide Separation

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Molten salt immersion technique has been tested with several Sr oxides, SrZrO_3 , SrMoO_4 and U_2SrO_y , and MgCl_2 based molten salts for the Sr nuclide separation. Reaction time, temperature, and salt composition were varied to effectively separate Sr in chloride forms. ICP-OES, XRD, and SEM analysis were conducted for the conversion efficiency and structure and morphology analysis. It is confirmed that all experiments of SrZrO_3 with MgCl_2 at 800°C for reaction time 5, 10, 20 hours showed higher conversion efficiency than 99% and in LiCl-KCl-MgCl_2 and NaCl-MgCl_2 molten salts at 500°C or 600°C , conversion efficiency higher than 97% was obtained. SrMoO_4 in MgCl_2 immersion experiments for 10 hours showed higher conversion efficiency than 99% when the molar ratio of salt/oxide powder is 7. U_2SrO_y was also tested with MgCl_2 molten salt at 800°C and higher efficiency than 99% and mainly MgUO_4 were produced as a reaction product.

Keywords: MgCl_2 based molten salt, Sr nuclide separation, Sr oxide phases, Molten salt immersion technique

Development of Trapping System for CO₂ Released From Spent Nuclear Fuel

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An accumulation of spent nuclear fuel (SNF) has brought a considerable interest due to its energy and environmental issue. To effectively manage SNF, a pyroprocessing is introduced to separate useful resources from the spent fuels and to manufacture suitable fuels. In head-end process of pyroprocessing, spent fuels are thermally treated to prepare UO₂ pellets, where various radioactive gases from SNFs are released during thermal treatment. Within these gases, C-14 as CO₂ form is a radioactive fission product which had a long half-life of 5,730 years and emits beta radiation of 0.156 MeV. Generally, current CO₂ capturing technologies include adsorption by solid materials, absorption by aqueous solutions, and membrane separation. Among these methods, absorption is an effective approach which traps CO₂ effectively and it is easy to operate at room temperature. In addition, it is highly recommended as immobilizing ¹⁴CO₂ as CaCO₃ formation due to the high thermal and chemical stability, and the relatively low solubility in water. Generally, a double alkali method has been proposed to capture low concentrated ¹⁴CO₂ from the stream. This method for CO₂ capture includes absorption process with NaOH solution and causticization using Ca(OH)₂. In this study, CO₂ emitted from SNF is captured using double alkali method, and the effects of operating conditions on capturing efficiency were investigated. Furthermore, considering the two-film theory, the effects of trapping conditions on the CO₂ absorption performance were examined. The recovered CaCO₃ from causticization was collected from the absorbing solution and analyzed.

Keywords: CO₂, Absorption, Double film theory, Spent fuel, Pyroprocessing

Consideration on Design Loads Needed for Assessing Integrity of Fuel Assemblies Under Normal Conditions of Transport

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This paper intends to present considerations on the question of what is the “load standard” or “design load” for integrity evaluation under normal transportation conditions and what type of design load is good for users. This suggests a direction for subsequent research on producing design loads that transport business companies can utilize without difficulty. Several studies have been conducted to evaluate the integrity of spent nuclear fuel during normal transportation. A representative study recently conducted is the Multi-modal Transportation Test (MMTT) conducted using a commercial spent nuclear fuel cask by US DOE in 2017. In Korea, additional transport tests were planned to acquire sufficient test data under the conditions of road and sea transport considering the Korean situation. As a result, road transport tests were carried out in 2020 and sea transport tests were carried out in 2021.

In the road transport test, a driving test that simulates various road conditions and a test that cycled a 4.5 km road eight times were performed. In most cases, the maximum acceleration of less than 1 g occurred, and the maximum strain was less than 48 $\mu\epsilon$. For the sea transport test, the magnitude of both the maximum acceleration and the maximum strain were lower than those in the road transport test. We concluded tentatively that the integrity of spent fuel under normal conditions of transport was satisfactory with a large margin. However, when the storage business is realized and the transport of spent fuel becomes visible, the storage and transport business companies will have to prove the maintenance of the integrity of the spent fuel under normal transport conditions at the request of the regulatory agency. The transport business companies can transport the spent nuclear fuel by using different types of transport casks and different types of trucks and ships from those used in the tests mentioned above. However, it is absurd to have to prove the integrity of spent nuclear fuel by performing expensive tests again.

Therefore, in this study, the design load that can be used by transport business companies is to be presented. The design load to be presented should satisfy the following requirements. The design load should be applicable including some differences in the transport cask or transport system, or different design loads should be presented according to the differences. The location where this design load is applied is to be specified (e.g. fuel rod, basket, internal structure). Requirements according to the operating speed of the transport system should be presented together. The type of design load is to be presented (e.g. PSD, SRS, FDS etc.). Other types of standards may be presented. For example, a speed limit for a vehicle carrying spent nuclear fuel may be suggested, or a speed limit for a vehicle passing through a speed bump may be suggested.

In order to present such a reliable design load, a multi-axis vibration excitation shaker table test will be carried out. Though this shaker table test, the behavior of the nuclear fuel assembly is closely evaluated by applying the data obtained from the road and sea transport tests previously performed as an input load. In addition, FDS (Fatigue Damage Spectrum) will be produced and applied to experimentally evaluate the durability of fuel assemblies under normal transport conditions.

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Keywords: Design load, Normal condition of transport, Spent nuclear fuel, Shaker table test

Fabrication and Preliminary Operation of Equipment for Formation of Bead-Type Sorbent to Capture Iodine

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The Na, Al, and Si akoxides-based sorbents for iodine capture have higher maximum iodine capture capacity and pore properties than zeolite-based sorbents. However, these sorbents were prepared in the form of granules via a step for cutting cylindrical alcogels. Since as-made sorbents decreased packing density, they must be additionally crushed and then classified into an appropriate size for increasing packing density. The bead formation in the step of sol-gelation could bring about the simplification of sorbent fabrication process and an improvement of packing density. For the formation of gel bead, characteristics such as hydrophilic or hydrophobic property and density of sol solution were investigated to design sol-gelation equipment. The sol-gel bead preparation equipment in the reflection of sol solution characteristics was fabricated through selection of oil for formation of sol bead, solvent for collection of gel bead, and nozzle for spray of sol droplet formation. The continuous or discontinuous formation of sol beads from NaAlSi-OH sol solution were observed according to flow rates of 6 to 8 mL·min⁻¹ and nozzle diameters of 0.4 to 0.8 mm. In the sphericity of sol bead, the best sol beads were obtained from 0.5 mm nozzle without clogging by sol solution in the flow rate range of 6–8·min⁻¹.

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

A Parameter Study on Specimen Geometry for Concrete Dynamic Compression Test

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During normal and off-normal conditions, the concrete structures of dry storage system for spent nuclear fuel must maintain structural integrity. A stress-strain curve is the most important key factor for structural integrity evaluation. The ASTM C39 specifies the concrete specimen geometry for the static compression test. However, there is no standard specimen size for intermediate strain rate, and it is not easy to maintain consistency among all test results because the failure tendency is different from each other. In order to account for the strain rate effects on concrete, the dynamic increase factor (DIF) is conventionally addressed by dividing dynamic strength by static strength. However, the DIF value considers only the strength of concrete and does not describe the overall behavior of concrete, such as a stress-strain relation. The objective of this study is to propose proper specimen geometry for the concrete dynamic compression test by several parametric study. The static compression simulation results with the specimen specified in ASTM C39 showed the constant strain distribution in a cylindrical specimen. However, as the strain rate increases, the strain state in specimen showed a non-uniform with the same geometry of ASTM C39. The non-uniform strain state in the specimen deteriorates the consistency and accuracy of the compression test. Therefore, we presented the specimen shape and size to form a uniform strain state through radial direction by drilling a hole in the axial direction. We analyzed two specimens using ABAQUS with the concrete damaged plasticity model, one with a hole at the center and the other without the hole. As a result, the strain distribution became more uniform than the specimen without the hole. Based on the results, we proposed the specimen shape and size for the intermediate strain rate compression test.

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Keywords: Concrete, Dry storage system, Strain rate, Specimen geometry

Collaborative R&D Work Between Korea and USA for Cooling Rate Effect of Hydride Reorientation

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Laboratory testing to simulate the drying of spent fuel is most often done using a cooling rate of approximately 5°C per hour because there are so many restricted test conditions like R&D project duration limit, budget and temporary electronic supply blackout at laboratory building. However, in a real dry cask storage system, the fuel cools much slower. Early data from KAERI on unirradiated, pre-hydrated cladding has shown that slower cooling may result in more brittle behavior than is currently observed based on these short-term tests. Given the potential safety and future handling implications of failed fuel, it is important to determine if the material properties of spent fuel cladding measured in these laboratory tests are the same as would be observed on fuel that has undergone a much longer, slower cooling, which may provide more time for hydrides to precipitate in the radial direction. KAERI and PNNL have started a collaborative I-NERI R&D project on this topic and each organization will perform tests on unirradiated & irradiated cladding under various hoop stress and cooling rate combinations. Scope of collaborative work is to evaluate long-term cooling (slow cooling rate) on hydride reorientation and subsequent material properties of cladding to determine if past and current research activities on spent nuclear fuel are bounding. The results will be used to direct future testing and help predict cladding performance over a wide range of burnups during extended storage and transportation.

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Keywords: Spent fuel, Hydride reorientation, Slow cooling rate, I-NERI

Subchannel Analysis of PWR Single Fuel Assembly Using COBRA-SFS

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Detailed temperature distributions of the spent fuel are required to evaluate the long-term integrity of the dry storage system. In this study, a subchannel analysis method was established to obtain the detailed temperatures of a spent fuel using the COBRA-SFS code.

The SAHTT (Single Assembly Heat Transfer Test) model was selected as the subchannel analysis. It was developed at the PNL to investigate heat transfer characteristics of spent PWR fuel under dry storage conditions. The SAHTT has a 15×15 rod array with simulated rods 0.42 in. (10.7 mm) in diameter. Control rod thimbles were modeled with unheated rods. The COBRA-SFS input consists a detailed subchannel model with 256 subchannels, 225 rods, and 8 slab nodes. The heat generation rate was axially uniform with total power of 1.0 kW.

Subchannel analyses were performed for the vertical orientation under three different backfills of air, helium, and vacuum. For the vacuum backfill, the peak temperature was the highest and temperature gradients the sharpest only due to the radiation heat transfer effect. For the helium backfill, peak temperature was lowest and the axial profiles flattest due to the higher conductivity and lower density of helium.

Subchannel analyses were also performed to evaluate the effect of thermal parameters such as surface emissivity, convective heat transfer coefficients, and flow resistance coefficients on the PCT (Peak Cladding Temperature). The PCT was affected by the emissivity of the fuel rod and the basket, and in particular, the basket emissivity had a greater effect. The PCT was affected by the Nusselt number, but the range of the Nusselt number is around 3.66. Therefore, the effect of the Nusselt number on the PCT will not be significant. As a result of the analysis according to the flow resistance coefficients, the PCT was affected by the wall friction factor, but the loss coefficients from the space grid had little effect. Subchannel technique obtained from this work can be used to predict the detailed temperature distributions of spent fuel assembly.

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Keywords: Subchannel analysis, Fuel assembly, Peak cladding temperature, Surface emissivity, Convective heat transfer, Flow resistance

Separation of Ce and Hf Using Liquid Bi Intermetallics Density Difference

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When disposing of spent nuclear fuel, there is a risk of exposure that could exceed the annual allowable dose due to human intrusion after the institutional control period. Therefore, it can be treated with the pyroprocess, but the decontamination factor is not sufficient, and an additional actinide recovery is required because molten waste salt-containing actinide is generated. In the case of reducing the element in the spent molten salt through an electrochemical method using a liquid Bi electrode, it is difficult to separate only the actinide element because the two-element groups are reduced together due to the large concentration difference between the actinide and the rare earth element. Therefore, a process of forming a Bi intermetallic compound using a liquid Bi electrode, which has higher element separation efficiency than a liquid Cd electrode, and physically separating the Bi intermetallic compound using the difference in density of the produced compound has been proposed. For this, it is necessary to understand the properties and density separation of the intermetallic compound to be produced, and experiments were planned and conducted for this purpose. Various metals were added to the molten Bi to form an intermetallic compound, and an analysis device such as SEM was used to determine the intermetallics distribution, composition, and internal structure. As the added metal, Ce is a representative element for lanthanide, and Hf with the most similar intermetallic density, decomposition temperature, and standard reduction potential to U, and U as a substitute element for actinide was adopted. As a result of SEM and EDS analysis, it was confirmed that the separation was made in Bi due to the density difference between the produced intermetallic compounds. A Ce-Bi intermetallic compound was observed in the upper part, Hf at a concentration smaller than the error range was detected, and a Hf-Bi intermetallic compound which containing high concentration of Ce was observed in the lower part. Separation of high-purity Ce seems to be possible in the upper part, and it seems difficult to separate high-purity Hf in the lower part. Therefore, to separate high-purity Hf, an additional process suitable for it seems to be necessary.

Keywords: Bi intermetallic, Density separation, Pyroprocess, Additional actinide recovery

FE Analysis on Drop Accident for Ship Loading Process of SF Transport Cask

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Some Spent Fuel Pools (SFPs) will be full of Spent Nuclear Fuels (SNFs) within several years. Because of this reason, transporting the SNF from SFP to interim storage facilities or permanent disposal facilities should be considered. There are two ways to transport the SNF from a site to other site, one is the land transportation with truck or train, and the other is the maritime transportation with ship. The maritime transportation has some advantages compared with the land transportation. The maritime transportation method uses safer route which is far from populated area than land transportation method, and transport more weight than land transportation method. However, the cask should be loaded into the ship for the maritime transportation, and there is a possibility of a drop accident of the cask onto the ship. Therefore, it is necessary to evaluate the structural integrity of the cask and ship for the drop accident during the loading process. To evaluate the structural integrity of the cask and ship, it is necessary to determine the analysis conditions that caused the greatest damage in the drop accident. There may be various conditions such as the drop angle of the cask, the initial falling speed, the drop position onto the ship, the size of the ship, etc. This study set the drop angle of the cask and the drop position onto the ship as the simulation variables, which have high possibility to occur during cask drop. However, the others are excluded since they are controllable by worker. In this paper, various drop angle (0, 15, 30, 45, and 70 degree) of the cask were simulated to define the greatest damage condition. KORAD-21 cask model was used for Finite Element Analysis (FEA), and FEA was performed to simulate a horizontal drop (1 m drop). The strain-hardening material properties for the deck were used as HT36 steel. The Cowper-Symonds constitutive model for HT36 was used to consider the strain rate effect. A Tie-down structure for supporting the cask was modeled with the cask model which contained inner structures like canister, basket, etc. Structural integrity of the cask and tie-down structure were evaluated using the von-Mises stress and equivalent plastic strain (PEEQ), and one of the ship deck was evaluated using deflection of ship deck and equivalent plastic strain. Compared with each cask drop angle conditions, 45 degree of the cask drop angle showed the highest deflection and PEEQ values, but did not exceed ultimate strain of HT36. In the ship deck, the corner of deck showed the highest PEEQ value in all simulation cases. As the result, the 45 degree of the cask drop angle condition results was more conservative than other conditions, and the corners of deck failure was able to evaluate ship safety.

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Keywords: Accident conditions of transport, Transport cask, Drop accident, Ship loading, Maritime transportation

Microstructure Analysis of Thermal Neutron Absorbing B₄C/Al Composites Fabricated by Using Stir Casting and Rolling Process

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In this study, for thermal neutron absorption, an aluminum metal composite in which B₄C particles were uniformly dispersed was prepared using stirring casting and hot rolling processes. The microstructure, thermal neutron absorption rate, mechanical properties and dispersibility of the reinforcement of the prepared B₄C/Al composite were analyzed. The composite in which the 40 μm sized B₄C particles were uniformly dispersed increased the tensile strength as the volume ratio of the reinforcement increased.

Keywords: Metal matrix composite, Boron carbide, Stir casting, Thermal neutron absorption

A Parametric Study of Confinement Evaluation of Spent Fuel Dry Storage Systems for Accident Condition

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In order to construct and operate the dry storage systems, it is essential to confirm the safety of the systems through safety analysis. If the dry storage cask is damaged due to an accident, a large amount of radioactive material may be leaked to the outside and cause radiation exposure to surrounding workers and nearby public, so the effect thereof should be evaluated. Many input parameter are required in the confinement evaluation for accident condition, and in this study, the change in the confinement evaluation result according to the change of major input parameter is to be studied.

In this study, we selected fractions of radioactive materials available for release from spent fuel, cooling time, and distance to exclusive area boundary as the major input parameter. In general, the release fraction suggested by NUREG-1536 has been used, but NUREG-2224 provides the fraction for high burn-up spent fuel in fire and impact accident conditions, unlike NUREG-1536 which provide a single value. In the case of the distance to exclusive area boundary, 100 to 800 m was considered, and in the case of the cooling time, 10 to 50 years was considered in this study. In order to compare the dose change by the parameter, we set up the hypothetical storage system. A storage cask of the system contain 21 PWR spent fuel assemblies with an initial enrichment of 4.5wt%, burnup of 45,000 MWD/MTU. During the accident condition, it is assumed that the cask is leaked at $1.0 \times 10^{-7} \text{ cm}^3 \cdot \text{sec}^{-1}$. Since the main dose criterion for accident conditions is 50 mSv of effective dose, effective doses are calculated in this study.

In an accident condition, transuranic particulate contribute most of the doses, so the doses are determined according to the fraction for the particulate. Therefore, it was confirmed that the dose was almost the same as the fraction for the accident conditions in NUREG-1536 and the fraction for the impact accident conditions in NUREG-2224 is 3×10^{-5} , but the dose was also 100 times higher as the fraction for the fire accident conditions in NUREG-2224 is 3×10^{-3} . In the case of the cooling time, it was confirmed that the dose change according to the cooling time was not significant because the dose contribution of transuranic elements having very long half-life was very large. In the case of the distance, it was confirmed that the dose decreased exponentially as the atmospheric dispersion factor decreased exponentially with the distance.

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Keywords: Confinement evaluation, Spent fuel, Dry storage, Parametric study, Accident condition, Effective dose

Option Strategies on the Pre-management Processes of Spent Nuclear Fuel for Waste Disposal System

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Korea Atomic Energy Research Institute (KAERI) has investigated Pyroprocessing technology in order to decrease the burden of disposal system and increase availability of useful radionuclides in the spent nuclear fuel (SNF) for future. The treatment and the disposal of SNF, however, are very sensitive issues socially. In addition, under the energy transition policy phasing out nuclear energy gradually there have been demands for alternatives so far. Thus various alternatives should need to be investigated in preparation for unexpected situations.

This study has been conducted roughly in effectiveness point of view of alternative pre-managements for SNF, not pyroprocessing technology, in disposal system, consisting of three stages according to the degree of burden in disposal system. Stage I is the case for making safety increase with removing highly-mobile radionuclides from SNF. Stage II is the case for eliminating high-heat radionuclides additionally, alleviating thermal risk in the disposal system. And Stage III is the case for recovering Uranium in addition to Stage II.

These options of pre-management are thought to be able to provide an intuitive strategy for effective diversification of the disposal system. Because several types of waste form from pre-management make it possible to develop the effective, newly-composed waste disposal system according to the properties of radionuclides. And the processability of SNF through pre-management might be combination with available core-drilling technology, being able to design various disposal system as well.

Even though the whole, detailed unit processes have not designed yet, mass balance and distributions of radionuclides are performed under the appropriate assumption of engineering processes. As a first step the alternative approaches for SNF pre-management for disposal system might be expected to be widely used in implementing SNF management policy in the future.

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Keywords: Option, Spent nuclear fuel, Pre-management, Waste disposal system

Condition of Corrosion Simulation for CANDU Spent Nuclear Fuel

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PWR spent nuclear fuel generally showed an oxide film thickness of 100 μm or more with a combustion rate of 45 MWD/MTU or higher, while CANDU spent nuclear fuel with an average combustion rate of about 7.8 MWD/MTU had few issues related to hydride corrosion. Even based on the actual power plant data, it is known that the thickness of the oxide film is 10 μm or less on the surface of the coating tube, and brittleness caused by hydride is shown from the thickness of the oxide film of about 80 μm , so it is not worth considering. However, since corrosion may be accelerated by lithium ions, lithium ions may be said to be a very important factor in controlling the hydro-chemical environment of heavy water. Lithium has a negative effect on the corrosion of zirconium alloys. However, since local below 5 ppb to prevent corrosion. maintained at a concentration between 0.35 and 0.55 ppm. Hydrogen is known to have a positive effect by suppressing radioactive decomposition of the coolant and suppressing cracks in nickelbased alloys. However, too much hydrogen can produce hydride in a pressure tube composed of Zr-2.5Nb, so DH (Dissolved Hydrogen) maintains the range of 0.27–0.90 ppm. pH and conductivity are completely determined by lithium ions, and DH can be completely removed below 5 ppb to prevent corrosion. Therefore, for cladding corrosion simulation of the CANDU spent nuclear fuel, a hydrochemical of the equipment, not 310°C, and 14 $\mu\text{S}\cdot\text{cm}^{-1}$ is targeted as conditions for corrosion acceleration. In addition, for acceleration, the temperature was set to 345°C (margin 10°C), which is the maximum accommodation range of the equipment, not 310°C.

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Keywords: Corrosion simulation/ acceleration, CANDU spent fuel, Lithium, Dissolved hydrogen

**~~Mechanical Integrity Evaluation on the Spent Nuclear Fuel Cladding Under
Axial and Bending Loads During the Transportation~~**

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Canceled Poster

Case Study on the Quantification of Residual Moisture in Dry Storage System

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To dry storage of spent nuclear fuel withdrawn the wet storage, all moisture inside the dry storage container must be removed to ensure the long-term integrity and retrievability. Substantial amounts of residual water in dry storage container may have potential impacts on the fuel, cladding, and other components in the dry storage system, such as fuel degradation and cladding corrosion, embrittlement, and breaching. The drying could perform as a vacuum drying process or a forced helium dehydration process. In NUREG-1536, the evacuation of most water contained within the canister is recommended a pressure of 0.4 kPa (3 torr) to be held in the canister for at least 30 minutes while isolated from active vacuum pumping as a measure of sufficient dryness in the canister. Monitoring the moisture content in gas removed from the canister is considered as a means of evaluating adequate dryness. Dew point monitoring and special techniques could be used to evaluate this adequacy. Various studies are continuing for quantitative evaluation of residual moisture inside the dry storage system. Andrawes proposed a methodology for determining trace water contents in gaseous mixtures, utilizing gas chromatography together with a helium ionization source. A microwave plasma source and emission spectrometry were utilized to determine trace amounts of bound water in solid samples using peak areas of atomic oxygen (O) and hydrogen (H) emissions. Bryans measured the gas samples taken from the High Burn-Up Demonstration Cask at three intervals: 5 hours, 5 days, and 12 days after the completion of drying and backfilling in the North Anna power Station. To measure water content, a Vaisala humidity probe was used. Final results indicated that the cask gas water content built up over 12 days to a value of 17,400 ppmv $\pm 10\%$, equivalent to approximately 100 g of water within the entire cask gas phase. Tahiyats also proposed a methodology that involves a direct current (dc) driven plasma discharge and optical emission spectroscopy for detecting and quantifying water vapor in a flowing gas stream under both trace and high water vapor loading conditions. For detecting water vapor concentration, the emission from H at 656.2 nm was employed. The H emission is the red visible spectral line generated by a hydrogen atom when an electron falls from the third lowest to the second lowest energy level, this suggests that the normalized H intensity can be used as a marker for water vapor detection and quantification. Several of the attempts are continuing to quantify water contents in dry storage system. Lessons learned by Case studies would be provided insights into how to improve future measurements.

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Keywords: Quantification, Residual moisture, Dry storage, Drying adequacy

A Study on the Applicability of Simplified Model of High Burnup Spent Fuel Rod Using Stress Correction Factor in Vertical and Horizontal Drop Impact

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For economic and safe management of Spent Nuclear Fuel (SNF), it is very important to maintain the structural integrity of SNF and to keep the fuel undamaged and handleable. The cladding surrounding nuclear fuel must be protected from physical and mechanical deterioration. The structural evaluation of SNF is very complicated and numerically demanding and it is essential to develop a simplified model for the fuel rod. In this study, a simplified model was developed using a new cladding failure criterion. The simplified model was developed considering only the horizontal or lateral static load utilizing the cladding material properties of irradiated Zircaloy-4, and applicability in horizontal and vertical drop impacts was investigated. When a fuel rod is subject to bending, a very complicated 3D stress state is generated within the vicinity of the pellet–pellet interface. A very localized stress concentration is observed in the area where the edges of the pellets contact the cladding. If the failure strain criteria obtained from the uniaxial tension test or biaxial tube test is applied, 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 criteria. In this work, a new cladding failure criterion is proposed, 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 failure criterion for SNF was selected as the membrane plus bending stress through stress linearization in the cross-sections through the thickness of the cladding. Because the stress concentration in the cladding around the vicinity of the pellet–pellet interface cannot be simulated in a simplified beam model, a stress correction factor is derived through a comparison of the simplified model and detailed model. The applicability of the developed simplified model is checked through horizontal and vertical drop impact simulations. It is shown that the stress correction factor derived considering static bending loading can be effectively applied to the dynamic impact analyses in both horizontal and vertical orientations.

Keywords: Spent nuclear fuel, Cladding, Simplified model, Impact analyses

A Study on the Barrier Effect Model for Radioactive Material Release Rate Estimation Into the Ocean

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Barrier effect model developed by CRIEPI is used for the estimation of rate of radioactive material release from a transport cask submerged in the ocean. If the containment boundary of cask is broken in an accident during maritime transportation, the sea water comes into the cask cavity and the leaching of radioactive material occurs. If the release of radioactive material thorough the opening of the containment boundary of cask is less than the leaching rate of the radioactive material inside the cask, then the release rate is controlled by the saturation limit of the sea water inside the cask cavity. In this study, the release rate estimation using the barrier effect model is compared with the model used in other codes, such as MARINRAD. And by parameter study, important factors that affect the releaser rate are identified and prioritized. It is shown that the gap generated in the containment boundary is the key parameter that determine the release rate of the radioactive material and the leaching rate is the dominant parameter to determine the saturation time of the cavity sea water.

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

A Study of Thermal Conductivity and Emissivity Behavior of Spent Fuel for Safety Evaluation in Long-term Dry Storage

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The conventional research trend on spent fuel was safety analysis based on mechanical perspective. Analysis of spent fuel cladding is based on the temperature of cladding and pressure inside cladding. To improve fuel cladding analysis, precise and accurate thermal safety evaluation is required. In this study a database which is about thermal conductivity and emissivity for the thermal modeling was established for a long-term safety analysis of spent fuel. As a result, we confirmed that the thermal conductivity of zirconium hydride was not accounted in conventional model such as FRAPCON and MATPRO. The conductivity of zirconium and its oxide was evaluated only as a function of temperature. However, the behavior of heat conductivity and emissivity is determined by the change of the material properties. The material properties depend on the microstructural characteristic. It can be seen that this conventional approach does not consider the microstructure change behavior according to vacuum drying process or burn-up induced degradation phenomena. To improve the thermal properties of spent nuclear fuel cladding, the measurement experiments of heat conduction and emissivity are required according to spent fuel experience and status such as the number of vacuum drying, cooling rate, burn up, hydrogen concentration and oxidation degree. In previous domestic reports and papers, we found that relative data between thermal properties and spent fuel experience and status does not exist. Recently, in order to understand the failure mechanism of hydrogen embrittlement, many studies have been conducted by accounting and spent fuel experience and status in a mechanical perspective. If microstructure information could be obtained from these studies, the modeling of thermal conductivity and emissivity will be possible indirectly. According to a recent abroad paper, it was confirmed that the thermal conductivity decreased by about 30% due to irradiation damage. The radiation damage effects on thermal conductivity also has not been studied in zirconium oxide and hydride. These un-revealed phenomena will be considered for the thermal safety model of spent fuel.

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Keywords: Conductivity, Emissivity, Vacuum drying, Zirconium, Hydride, Spent nuclear fuel

Fine Powders Removal/Recover System Development for Porous U_3O_8 Pellets in Head-End

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Dry head end process is developing for pyro-processing at KAERI (Korea Atomic Energy Research Institute). Dry processes, which include disassembly, mechanical decladding, vol-oxidation, blending, compaction, and sintering shall be performed in advance as the head-end process of pyroprocessing. An important goal of the head-end process is the fabrication of a proper feed material for the subsequent electrolytic reduction process. In the vol-oxidation process, the pellet type-SFs are pulverized by an oxidation under an air-blowing condition, and some volatile fission products are removed from the produced powders by using an air flow. After blending, the U_3O_8 powders are moved to a compactor of compaction process to obtain U_3O_8 porous pellets. In the fine powders removal system connected with compactor, for the improved performance of oxide reduction process coupled to dry head-end process, the removal/recovery system for fine powders potentially attached to the surface of oxide reduction raw material was developed and applied to the removal of fine powders from green pellets fabricated in dry head-end process. The removal efficiency of fine powders was also verified using porous U_3O_8 pellets in the fine powders removal system.

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Keywords: Head-end, Fine, Powders, Removal, Pellets, U_3O_8

Effect of Oxygen Content on Fission Product Release Behavior

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In KAERI, the nuclide management technology is currently being developed for the reduction of disposal area required for spent fuel management. Among the all fission products of interest, Cs, I, Kr, Tc are considered to be significantly removed by following mid-temperature and high-temperature treatment, however, a difficulty of spent-fuel thermal treatment experiment limits the development of such thermal treatment. In this study, we applied our previously developed two-stage diffusion release model coupled to UO_2 oxidation model to the development of optima thermal treatment scenario. Since the formation of cesium pertechnetate should be avoided and the fission release behavior is considerably affected by the extent of oxygen, we obtained oxygen-content dependent model parameters for two-stage fission release model and applied the model to the evaluation of fission release behavior to different oxygen content and thermal treatment procedure. 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 and the thermal treatment condition reducing the chemical reaction between cesium and technetium could be developed.

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Keywords: Two stage-diffusion model, Fission product release, Diffusion coefficient, Oxygen content, One-dimensional

3분과

고준위폐기물(Oral)

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

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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 assessments of 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. Geological development processes that the site have been 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. With regard to disposal site, understanding past geological evolution history allows to access the possibility of hazardous events of the site that can cause disturbances to the facility over the period of interest, and to verify the change in the geological environment is within the safe performance range even after the period of interest. In addition, certain parameters that change with the geological evolution can affect the hydrological and geochemical characteristics which are essential to disposal performance. There are various factors in the evolution of the geological environment, but not all are related to disposal safety. The objective of this research is to develop a geological reconstruction method considering factors that should be derived preferentially for the geological characteristics of the disposal site and the evaluation of the long-term safety. As a preliminary study on this, we investigated case studies related to geological reconstruction of overseas disposal research institutes, and reviewed which factors are suitable for the domestic granitoid distribution environment. It is expected that systematic and consistent results will be possible in the future through this methodology.

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

Preliminary Study on the Long-term Structural Evolution of Deep Geological Repository Site

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Deep geological disposal with multiple barriers composed of engineered and natural barriers has been considered as the most suitable disposal method for high level nuclear wastes. In terms of the geological evaluation factors, brittle structures such as fractures and faults should be characterized around the repository site, because radionuclides transfer mainly with groundwater in the subsurface and groundwater flows through discontinuous brittle structures. The geological survey for the characterization of deep geological repository sites is widely conducted by narrowing the survey area from regional scale down to local scale, which could be divided into three steps: 1) using remote sense or geophysical survey, 2) trench and drill core logging including field survey based on the first step, 3) detailed geological survey in the tunnel. In this study, we analyzed the distribution of geological structures to derive the history of brittle deformation in and around the KURT (KAERI Underground Research Tunnel) site located in the KAERI (Korea Atomic Energy Research Institute). The bedrock of the KURT site is mainly consist of Jurassic two-mica granite, which is extensively intruded by andesitic dikes of Cretaceous with N-S to NE-SW strikes. The two-mica granite in the study area was deformed in a ductile deformation environment and has been overprinted by major geological structures such as faults, dikes, veins, and joints. From this study, we identified 8 brittle deformation events based on the cross-cutting relationship among the geological structures, which are obtained from the analyses in and around the KURT. In order to evaluate the reactivation and fluid flow potential of brittle structures, it is essential to determine the characteristics and ages of the brittle structures and the composed rocks around the site.

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Keywords: KURT, Deep geological repository Site, Brittle deformation history, Long-term structural evolution

Age-dating Methods to Reconstruct the Long-term Evolutionary Processes 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 surface geological outcrop, tunnel face and drill core. 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 investigated proper age dating methods to construct lithological model of the disposal site with regard to the long-term safety. Also, preliminary age dating locations were selected using the lithological distribution results by depth through geochemical and micro-structural analysis for the deep drill cores excavated around KURT. In the study area, the dikes presumed the Cretaceous were intruded by Jurassic granites. As for the granotoids, U-Pb age dating for zircon, which is resistant to deformation or metamorphism and has loss, is often used. In the case of the dikes, K-Ar and $^{40}\text{Ar}/^{39}\text{Ar}$ age dating for the argon captured in the rocks after magmatism is often used. Through U-Pb zircon ages of KURT site granotoids, we expect to solve the clustering problem (granite and granodiorite), which is different from precious chemical analysis (XRF) results and TAS-diagrams. $^{40}\text{Ar}/^{39}\text{Ar}$ age dating to be used for the dikes is suitable for the perspective of lithological model of the disposal site. Because, it can compensate for accuracy problems such as sample heterogeneity in K-Ar age dating and is used for volcanic rocks. In the further study, we plan to determine the appropriate sampling locations by the selected age dating methods from the perspective of disposal in this study.

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Keywords: Geochronology, Disposal site, Lithological model, KURT, Natural barrier

History of Natural Barrier During Geological Time Scale, Using Paleomagnetism

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The radioactive waste repository consists of an engineered barrier and a natural barrier and must be managed safely after isolation. We classify the geological events of natural barriers for the evaluation of their present and future disposal stability assessment, they can be divided into regional and regional evolutions according to their scale. Regional evolution can be quantitatively explained by plate tectonics and regional rock distribution, and local evolution can be explained by petrological, mineralogical evidence and ductile, brittle deformation. Plate tectonics can explain the change quantitatively by restoring the direction of the Earth's magnetic field recorded when rocks were formed. The time units for these changes are tens of millions of years to hundreds of millions of years, but plate tectonic is a way to estimate geological history. It can be assessed by extrapolating past knowledge considering the known geological events of radioactive waste repository. It is possible to derive a conservative value of the change of the geological environment in the time unit of disposal stability. The Korean Peninsula belongs to the edge of the Eurasian plate and is divided into Gyeonggi, Yeongnam Massif, Okcheon orogeny belt, and Gyeongsang Basin. To quantitatively determine their geological history, we collected paleomagnetic data using rocks from the Korea Peninsula (paleomagnetic database and papers). We attempted to carry out the apparent polar wander paths (APWPs) on the Korean Peninsula by collecting and sorting data. Since the Korean Peninsula is composed of multiple massifs, this APWP is expected to serve as a basis for explaining the local crustal rotation or brittle ductile deformation. Furthermore, by extrapolating the change pattern from the past to the present, it can contribute to the estimation of the future geological evolution.

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Keywords: Natural barrier, Paleomagnetism, Geological history, Plate tectonics

Review of Classification System and Key Elements of NEA International FEP List

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Comprehensive identification and systematic classification of all features, events and processes (FEP) that influence on the performance of a high-level radioactive waste disposal system is essential for safety assessment. Nuclear energy agency (NEA) has been developing and updating the standardized generic FEP list, so-called NEA international FEP list, which may be used as the basis to develop project-specific FEP lists to reflect diverse system and site characteristics in different countries. On the basis, Finland and Sweden have recently got licenses to construct spent nuclear fuel deep disposal facilities. Also in Korea, timely construction of a high-level radioactive waste disposal facility is an urgent issue for stable operation of nuclear power plants. For this end, a FEP list that properly considers for system and site characteristics of Korean high-level radioactive waste disposal facility needs to be developed. In this study, the most recent NEA international FEP list published in 2019 was comprehensively reviewed with focus on the structure of the classification system and the physicochemical mechanisms associated with the key elements. The obtained results will be used for the comparative analysis of domestic and overseas project-specific FEP lists and for the development of a generic FEP list relevant to Korean high-level radioactive waste disposal system.

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Keywords: Spent nuclear fuel, Disposal, Safety assessment, FEP, Classification system

A Database Design Study for FEP Management of Korean Spent Nuclear Fuel Repository

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Since July 2021, the Korea Radioactive Waste Agency has been conducting a safety case development study for the Korean deep geological repository program. The safety case includes generating scenarios in which radioactive materials from spent nuclear fuel repository reach the human biosphere by combining selective FEPs (Features, Events, and Processes). This safety case should be able to transparently explain the process in which conclusions have been drawn not only to stakeholders but also to the public by presenting safety arguments. The scenario development stage consisting of FEP screening, scenario generation, and uncertainty analysis procedures should have a database management system. Database management system was performed in countries such as Sweden, which obtained approval for the construction of spent nuclear fuel repositories, and the United States, where various preliminary research was carried out. Korea Atomic Energy Research Institute also has experience in designing and operating its own database, which has conducted preliminary research on disposal of the spent nuclear fuel. Currently, the safety assessment of the Korean spent nuclear fuel repository is in the early stages of research, but it is necessary to set up a basic framework for database design while the collection of FEP data from domestic and international preliminary studies is under development, and it is advantageous for efficient database construction and operation. Therefore, this paper presents the current status of database design considering completeness and transparency from the FEP screening stage to the scenario development stage in the safety assessment process of the Korean spent nuclear fuel repository. In this process, the functional requirements that the database should provide, the database schema capable of implementing them, and simple examples are presented together. The objectives of this database design are flexible FEPs management, high integrity and consistency, and expandability for linking with the safety case database. The FEP data to be inputted into the database includes a list of major opened FEPs, including International FEPs from Nuclear Energy Agency, which were referred for PFEPs (Project-specific FEPs), and PFEPs applied to POSIVA's Olkiluoto repository. As an additional function, queries from the database are used to visually express the process of deriving scenarios through Rock Engineering System, a widely known scenario generation methodology.

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Keywords: FEP, Scenario, Database, Spent nuclear fuel repository

Effect of Damage Progress and Loading Type on Acoustic Emission Characteristics in Concrete for Geological Disposal of High-Level Nuclear Wastes

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Deep geologic disposal of high-level nuclear wastes (HLW) requires intensive monitoring instrumentations to ensure long-term security. Acoustic emission (AE) method is considered as an effective method to monitor the mechanical degradation of natural rock and man-made concrete structures. The objectives of this study are (a) to identify the AE characteristics emitted from concretes as concrete materials under different types of loading, (b) to suggest AE parametric criteria to determine loading types and estimate the failure stage, and finally (c) to examine the feasibility of using AE method for real-time monitoring of geologic disposal system of HLW. This study performs a series of the mechanical experiments on concrete samples simultaneously with AE monitoring, including the uniaxial compression test (UCT), Brazilian tensile test (BTT) and punch through shear test (PTST). These mechanical tests are chosen to explore the effect of loading types on the resulting AE characteristics. This study selects important AE parameters which includes the AE count, average frequency (AF) and RA value in the time domain, and the peak frequency (PF) and centroid frequency in the frequency domain. The result reveals that the cumulative AE counts, the maximum RA value and the moving average PF show their potentials as indicators to damage progress for a certain loading type. The observed trends in the cumulative AE counts and the maximum RA value show three unique stages with an increase in applied stress: the steady state stage (or crack initiation stage; < 70% of yield stress), the transition stage (or damage progression stage; 70–90% of yield stress) and the rising stage (or failure stage; > 90% of yield stress). In addition, the moving average PF of PTST in the early damage stage appears to be particularly lower than that of UCT and BTT. The loading in BTT renders distinctive responses in the slope of the maximum RA–cumulative AE count (or $\tan \theta$). The slope value shows less than 0.25 when the stress is close to 30% of BTT, 60% of UCT and 75% of PTST and mostly after 90% of yield stress, the slope mostly decreases than 0.25 in all tests. This study advances our understanding on AE responses of concrete materials with well-controlled laboratory-scale experimental AE data, and provides insights into further development of AE-base real-time diagnostic monitoring of structures made of rocks and concretes.

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Keywords: High-level radioactive nuclear waste, Acoustic emission, Concrete, Loading type, Damage

Natural Analogue Studies Using Uranium Isotopes and Their Ratios

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Uranium isotopes (^{238}U , ^{235}U , and ^{234}U) found in natural environments and their activity ratios ($^{235}\text{U}/^{238}\text{U}$ and $^{234}\text{U}/^{238}\text{U}$) have been used as an important tool in investigating various geological processes, especially in natural analogue studies. Occurrence and fractionation of uranium isotopes in nature between ^{238}U , ^{235}U , and ^{234}U were investigated. Various measurement methods have been used for the determination of isotopic ratios and geochronology. Thus, we reviewed and summarized the measurement methods such as alpha spectrometry, gamma spectrometry, thermal ionization mass spectrometry (TIMS), secondary ion mass spectrometry (SIMS) with sensitive high resolution ion microprobe (SHRIMP), and multiple-collector inductively coupled plasma mass spectrometry (MC-ICPMS).

Research status of natural analogue studies carried out using uranium isotopes and their isotopic ratios were also reviewed and summarized in terms of long-term behaviors of radionuclides in various foreign uranium deposits as analogues of high-level radioactive waste repositories. Research results for mineralogical, geochemical, and biogeochemical behaviors of uranium in various natural analogue sites were collected and analyzed to investigate the migration and retardation processes of uranium through geological media. These long-term behaviors of uranium and uranium isotopes include dissolution/precipitation of uranium minerals, interactions of uranium with various fracture minerals including sorption and incorporation, redox reactions by minerals and microbes, and hydrological groundwater flow thorough rock fracture systems including identification of flow paths and groundwater circulation.

The results of this study will contribute to performing future natural analogue studies in domestic uranium deposits and provide basic information and knowledge for understanding long-term geochemical and geochronological behaviors of radionuclides in a high-level radioactive repository.

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Keywords: Natural analogue study, Uranium isotope, Activity ratio, Isotope fractionation, Uranium deposit

Application of Thermodynamic Sorption Modelling: Separation and Characterization of Bentonil-WRK Montmorillonite

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A deep geological disposal system, which consists of the engineered and natural barrier components, is the most proven and widely adopted concept for a permanent disposal of the high level radioactive waste (HLW) thus far. The clay-based engineered barrier is designed to not only absorb mechanical stress caused by the geological activities, but also prevent inflow of groundwater to canister and outflow of radionuclides by providing abundant sorption sites. The principal mineralogical constituent of the clay material is montmorillonite, which is a 2:1 phyllosilicate having two tetrahedral sheets of SiO_2 sandwiching an octahedral sheet of Al_2O_3 . The stacking of SiO_2 and Al_2O_3 sheets form the layered structures, and ion-exchange and water uptake reactions occur in the interlayer space. In order to reliably assess the radionuclide retention capacity of engineered barrier under wide geochemical conditions relevant to the geological disposal environments, sorption mechanisms between montmorillonite and radionuclides should be explicitly investigated in advance.

Thus far, sorption behavior of mineral adsorbents with radionuclides has been quantified by the sorption-desorption distribution coefficient (K_d), which is simply defined as the ratio of radionuclide concentration in the solid phase to that in the equilibrium solution; the K_d value is conditional, and there have been scientific efforts to develop geochemically robust bases for parameterizing the sorption phenomena more reliably. In this framework, application of thermodynamic sorption model (TSM), which is theoretically based on the concept of widely accepted equilibrium models for aquatic chemistry, offers the potential to improve confidence in demonstration of radionuclide sorption reactions on the mineral adsorbents. Specifically, it is generally regarded in the TSM that coordination of radionuclides on montmorillonite takes place at the surficial aluminol and silanol groups while their ion-exchange reactions occur in the interlayer space also. The effects of electrical charge on the surface reactions are additionally corrected in accordance with the numerous theories of electrochemical interface.

The present work provides an overview of the current status of application of TSM for quantifying sorption behaviors of radionuclides on montmorillonite and experimental results for physical separation and characterization of Ca-montmorillonite from the newly adopted reference bentonite (Bentonil-WRK) by means of XRD, BET, FTIR, CEC measurement, and acid-base titration. The determined mineralogical and chemical properties of the montmorillonite obtained will be used as input parameters for further sorption studies of radionuclides with the Bentonil-WRK montmorillonite.

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Keywords: Radionuclide migration, Buffer, Tunnel backfill, Montmorillonite, Sorption, TSM

Optimization of Sorption Capacities and Affinities on the Multiple Sorptive Sites for Natural Illitic Clays

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Multiple sorptive sites on natural illitic clays (e.g., frayed edge [FES], type II [TS], and planar sites [PS]) play an important role to diverse ¹³⁷Cs immobilization in soil and aquifer environments. This study investigated the Cs sorption capabilities of 10 natural illitic clays at ranged Cs concentrations (1×10^{-7} to 1×10^{-3} mol·L⁻¹) under various competing potassium concentration (distilled water to 1×10^{-1} mol·L⁻¹). Additionally, multisite cation exchange model was performed to evaluate the best-fit sorption model and optimize the sorption capacities and affinities of multiple sorptive sites for Cs. Here, the experimental Cs sorption isotherms varied among 10 illitic clays, indicating different sorption capacities of Cs on illitic clays. The best-fit sorption model exhibited that variable Cs sorption of 10 illitic clays was significantly related to the sorption capacities at the FES (1.76×10^{-5} to 1.12×10^{-4} eq·kg⁻¹), TS (1.59×10^{-3} to 9.76×10^{-3} eq·kg⁻¹), and PS (2.14×10^{-2} to 1.51×10^{-1} eq·kg⁻¹), respectively. The FES predominantly contributed to Cs sorption at low aqueous concentrations, whereas the TS and PS sorbed Cs at high concentrations. These sorption capabilities of multiple sorptive sites were correlated to illite contents and crystallinity of illitic clays, implicating that such parameters could be key factors to predict the Cs sorption for natural illitic clays in soil and aquifer environments. Finally, 1-D transport simulations represented that the severe Cs retardation occurred at low Cs concentration, implying that the FES predominantly affected to Cs transport in actual radioactive contamination sites (i.e., where low Cs concentration prevails), compared to the TS and/or PS.

Keywords: Multisite cation exchange model, Illitic clay, Sorption, Radiocesium

Transient Mixed-potential Model for Oxidic and Anoxic Corrosion of a Cu Canister

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Corrosion of copper (Cu) canisters is one of the important factors to ensure the safety of a deep geological repository site. This is because the corrosion of a canister may induce failure of the canister which can lead to a release of radionuclides into the environment. Corrosion of canisters for high-level wastes is affected by the following multiphysics: thermal-hydraulics, transportation of chemical species, chemical reactions, and interface reactions. This research aimed to develop a multiphysics numerical model for the corrosion of spent nuclear fuel canisters for a deep geological repository in South Korea. The multiphysics model is based on MOOSE (Multiphysics Object-Oriented Simulation Environment) which uses a finite element method. In the multiphysics model, the following multiphysics are coupled and solved together for a deep geological repository design of South Korea: interface redox reactions, porous flow, and heat transport in porous flow. The proposed model was validated with experimental data before being applied to a KAERI reference disposal unit. It was found that the corrosion potential of a Cu canister shows an uneven distribution of corrosion potential along with the surface. In addition, top, bottom, and side surfaces of the canisters show a different lifetime and corrosion potential. Important redox reactions for corrosion are changed along with time from a reduction of O_2 and anodic dissolution of Cu by Cl^- to sulfidation of Cu and reduction of water. The proposed model will be coupled with some important chemical reactions in engineering buffers and will be the base for the understanding of the behavior of Cu canisters in the KAERI reference disposal unit.

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Keywords: Mixed potential, Corrosion, Copper, Canister, High level waste, Finite element method

Estimation of Hydrogen Gas Production Rate According to the Material of the Disposal Canister and the Disposal Environment in the Deep Disposal System

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The International Atomic Energy Agency recommends the deep geological disposal system as one of the disposal methods for high-level radioactive waste (HLW), such as spent nuclear fuel. The deep geological disposal system disposes of HLW in a deep and stable geological formation to isolate the HLW from the human biosphere and restrict the inflow of radionuclides into the ecosystem. It mainly consists of an engineered barrier and a natural barrier. Safety evaluation using a numerical model has been performed primarily to evaluate the buffer's long-term stability. However, although the gas generation rate input for long-term stability evaluation is the critical factor that has the most significant influence on the long-term hydraulic-mechanical behavior of the buffer, in-depth research and experimental data are lacking. In this study, the gas generation rate on the interface between the disposal canister and the buffer material, a component of the engineered barrier, was mainly studied. Gas can be generated between the disposal canister and the buffer material due to various causes such as anaerobic corrosion of the disposal canister metal, organic matter decomposition, radiation decomposition, and steam generation due to high temperature. The generation of gas in such a disposal environment increases the pore gas pressure in the buffer and causes internal cracks. The occurred cracks increase the intrinsic permeability of the buffer, which leads to a decrease in the primary performance of the buffer. For this reason, it is essential to apply the appropriate gas generation rate according to the disposal condition and buffer material for accurate long-term stability analysis. Therefore, the theoretical models regarding the estimation of gas generation were summarized through a literature study. The amount of gas generated was estimated according to the disposal environment and material of the disposal canister. It is expected that estimated values might be used to estimate the long-term stability analysis of buffer performance according to the disposal condition.

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Keywords: Gas generation rate, Deep geological disposal system, Long-term stability of buffer, Interface between canister and buffer

Development and Application of Integrated Hydrologic Simulators for the Long-term Safety Analysis in the Deep Geologic Repository Technology Program

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Deep geologic repositories (DGR) are designed to store spent nuclear fuel and to isolate it from the biosphere for an extended period of time as long as millions of years. The long-term performance of the DGR relies on the performance of the natural geologic barriers after the end of the lifetime for the engineered barrier systems. Typically, multiple analytical and numerical models are used to analyze and ensure the safety of the repositories along both engineered and natural barrier systems. Despite the immense advancement in computing power and modeling techniques over the last few decades, a series of models and their linkage often require many simplifying assumptions in this safety assessment. The degree of the reliability and confidence of the safety analysis is thus highly dependent on the validity of those tools used. Considering the significance of the DGR performance and public attention, the highest level of quality control is necessary for the models employed in the assessment. The performance of the ultimate long-term geologic barrier is determined by the expected travel time of the radioactive species of interest, the level of their dilution or radioactivity at compliance areas, and the uncertainty associated with them. As the species of interest can be carried away from the repository location by groundwater flow, the travel time is determined by groundwater velocity along the flow path from source to biosphere while the dilution is a function of the decay and production rates as well as the diffusion and dispersion. Due to the time scale and the complexity of the physicochemical processes and geologic media involved, the models used for safety evaluation will need to become more and more comprehensive, robust, and efficient which is difficult to achieve in principle. They will also need to be transparent and flexible to satisfy the regulatory quality control requirements. This study thus attempts to develop an accessible, transparent, and extensible integrated hydrologic models (IHM) which can be widely accepted by the regulators as well as scientific community and thus suitable for current and future safety assessment of the DGR systems. The IHM can be considered as a tool and a framework at the same time when it is designed to easily accommodate additional processes and requirements for the future as it is necessary. The IHM is capable of handling the atmospheric, land surface, and subsurface processes for simultaneously analyzing the regional groundwater driving force and deep subsurface flow, and repository scale safety features, providing an ultimate basis for seamless safety assessment in the DGR program. The applicability of the IHM to the DGR safety assessment is demonstrated using simple illustrative examples.

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Keywords: Deep geologic repository (DGR), Safety assessment, Natural barrier, Integrated hydrologic model (IHM), Quality control requirement

Evaluation of Physicochemical Characteristics and Estimation of Centerline Temperature of Sr Ceramic Waste Form for Long-term Storage and Disposal

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To reduce the environmental burden caused by the disposal of spent nuclear fuel and maximize the utilization of the repository facility, waste burden minimization technology is currently being developed at the Korea Atomic Energy Research Institute (KAERI). The technology includes a nuclide management process that can maximize disposal efficiency by selectively separating and collecting major nuclides in spent nuclear fuel. In addition, for efficient storage facility utilization, the short-term decay heat generated by spent nuclear fuel must be removed from the waste stream. To minimize the short-term thermal load on the repository facility, it is necessary to separate heat generating nuclides such as Cs-137 and Sr-90 from the spent fuel. In particular, Sr-90 must be separated because it generates high heat during the decay process. KAERI has developed a technology for separating Sr nuclides from Group II nuclides separated through the nuclide management process. In this study, we prepared Sr ceramic waste form, SrTiO₃, by using the solid-state reaction method for long-term storage for the decay of separated Sr nuclides and evaluated the physicochemical properties of the waste form. Also, the radiological and thermal characteristics of the Sr waste form were evaluated by predicting the composition of Sr nuclides separated through the nuclide management process, and the estimation of centerline temperature was carried out using the experimental thermal data and steady state conduction equation in a long and solid cylinder type waste form. These results provided fundamental data for long-term storage and management of Sr waste.

Keywords: Strontium-90, Long-term storage, Ceramic waste form, Centerline temperature

Size Effect of Numerical Simulation on the Evolution of Fracture Aperture Around a Small Disposal Hole

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Through constructing statistical fracture network model based on discrete element method, the evolution characteristics of the fracture aperture had been directly simulated and evaluated caused by redistributed stress after the borehole excavation. This study focuses on the size effect of the discrete element method for the analysis of the effective distance of fracture aperture change after the borehole excavation. A two-dimensional trace-type domain with a maximum size of 1.1 m^2 was created using a discrete fracture network with stochastic information of KURT. A total of eight domains with different sizes were constructed from the largest domain area to the 0.4 m^2 analysis area. The aperture change ratio which can be depending on the domain size was examined. The ratio was investigated by comparing the aperture size before and after the simulation of borehole excavation. In addition, the effective range of aperture changes was analyzed by comparing the re-distribution distance from the center of the borehole. Based on dimensional analysis, input variables (borehole radius, occurrence distance of aperture changes, domain size) were modeled using exponential distribution form. Through the analysis model, two dimensionless variables were derived to investigate the expected distance of the aperture changes and appropriate DFN domain size for simulating bore excavation. As an application example of the 3-inch borehole simulation, the analysis model predicted that the range of aperture changes could occur within a radius of about 0.98 m from the borehole center, and the suitable size of the model had been inferred as about $5 \times 5 \text{ m}$ for minimizing the domain size effect.

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Keywords: Size effect, Small disposal hole, Discrete fracture network, Discrete element method

Experimental Investigation of Local Thermal Equilibrium Assumption Under Natural Conditions

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The mechanical, hydraulic, thermal, and chemical properties of the subsurface can have a significant effect on the long-term performance of an underground facility. Therefore, it is important to accurately estimate the aquifer properties in order to predict the groundwater flow and solute transport and thus ensure the stability and safety of a high-level radioactive waste disposal. Using heat as a tracer has become a popular tool for the subsurface characterization. Recent studies have demonstrated that heat tracing is an effective approach to quantify both hydrogeological and thermal subsurface properties. However, most studies in natural conditions assume the local thermal equilibrium (LTE) between the solid and fluid phases, ignoring heat exchange between them. The LTE assumption has not yet been verified by experiments. This work investigates the validity of the LTE assumption by performing the laboratory tracer tests using both solute and heat in a porous medium under natural groundwater flow velocities (Reynolds number, $Re < 0.37$). The experimental results showed that the LTE assumption can be violated even under natural groundwater flow conditions. The violation of LTE (LTNE) had a significant impact on mechanical dispersion, whereas its effect on velocity was negligible. These results provide the first experimental evidence for LTNE effects in natural conditions. Therefore, it is necessary to consider LTNE effects especially when the mechanical dispersion is evaluated using heat tracing.

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Keywords: Tracer test, Solute, Heat, Aquifer property, Local thermal equilibrium

3분과

고준위폐기물(Poster)

Is Surface Diffusion an Important Transport Mechanism in Rocks?

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Various diffusion experiments using geologic media have been carried out and it is often assumed that aqueous diffusion is the dominant transport mechanism. However, in some cases diffusive migration has been much faster than predicted in the model simulation. To explain such results surface diffusion of sorbing species was invoked. Experimental results were generally open to interpretation but possible existence of surface diffusion, whereby sorbed radionuclides could potentially migrate at much enhanced rates, necessitated investigation.

The potential for surface diffusion of some sorbing nuclides on through-diffusion experiments using domestic rocks was examined. The apparent diffusion coefficients for sorbing cations were determined from their steady-state diffusion flux through rocks disks, while effective and pore diffusion coefficients were obtained with non-sorbing tracers through the same rocks.

Diffusive transport models through domestic granites and granodiorites based only on pore diffusion did not often described adequately for sorbing cations. Thus, surface diffusion should be considered. Then what was the most important measure to estimate surface diffusion? As far as we examine, the sorption reversibility provides a hint of surface diffusion. The reversible sorbing species, for example, Sr, has a remarkable surface diffusion contribution, whereas surface diffusion has a relatively small contribution for irreversibly sorbing species such as Cs and Am under domestic experimental conditions.

Keywords: Diffusion, Sorption, Surface diffusion, Sorption reversibility, Sorbing species

THM Coupled Behavior of FE Experiment in Task C of the DECOVALEX-2023

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Geologic disposal of high-level radioactive waste is considered the most effective method to isolate high-level radioactive waste from the biosphere. A high-level radioactive waste repository is designed to be placed at a deep depth and generally consists of canisters, buffer material, and host rock. In the disposal system, the heat from the canister occurs for millions of years due to the long half-life of the high-level radioactive waste, and the heat induces vaporization of groundwater in the buffer material. The resaturation process also occurs due to groundwater inflow from the host rock by the hydraulic head and capillarity. The saturation variation leads to the heat transfer and multi-phase flow in the buffer material, and thermal pressurization of groundwater due to the heat affects the effective stress change in the host rock. The stress change can make the porosity and permeability change in the flow system of the host rock, and the flow system affects the nuclide migration to the biosphere. Therefore, it is crucial to understand the complex thermo-hydro-mechanical-chemical (THMC) coupled behavior to secure the repository's long-term safety.

DECOVALEX is an international cooperating project to develop numerical methods and models for predicting the THMC interactions in the disposal systems through validation and comparison with test results. In Task C of DECOVALEX-2023, nine participating groups (BGR, BGE, CAS, ENSI, GRS, KAERI, LBNL, NWMO, Sandia) models the full-scale emplacement (FE) experiments at the Mont Terri underground rock laboratory and focus on understanding pore pressure development, heat transfer, thermal pressurization, vaporization and resaturation process in the disposal system. In the FE experiment, three heaters generated heat with constant power for five years at a 1:1 scale in the emplacement tunnel based on Nagra's reference repository design. KAERI used OGS-FLAC3D for the numerical simulation, combining OpenGeoSys for TH simulation and FLAC3D for M simulation.

We generated a full-scale three-dimensional numerical model with a dimension of 100 by 100 by 60 meters. The pressure and temperature distribution were well simulated with the host rock's anisotropy. Based on the capillarity, we observed vaporization and resaturation in the bentonite under the two-phase flow system. We plan to compare the simulation results with the field data and investigate the effect of input parameters, including thermal conductivity and pore compressibility affecting the thermal and flow system.

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Keywords: DECOVALEX-2023, FE experiment, Mont-Terri URL, OGS-FLAC3D

Preliminary Study on Fluid Flow and Radionuclide Transport in Generic Repository Located at Fractured Crystalline Rock

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With the increase of temporarily-stored spent radioactive fuels, there is an increasing necessity for the safe disposal of high-level radioactive waste (HLW). Among various methods for the disposal of HLW, a deep geological disposal system is adapted as a HLW disposal strategy in many countries. Before the construction of a repository in deep geological condition, a performance assessment, which means the use of numerical models to simulate the long-term behavior of a multi-barrier system in HLW repository, has been widely performed to ensure the isolation of radionuclides from human and related environments for more than a million years.

Meanwhile, Korea Atomic Energy Research Institute (KAERI) is developing a process-based total system performance assessment framework for a geological disposal system (APro). To improve the reliability of APro, KAERI is participating in DECOVALEX-2023 Task F, which is the international joint program for the comparison of the models and methods used in deep geological performance assessment. As a final goal of Task F, the reference case for a generic repository in fractured crystalline rock is described. The three-dimensional generic repository is located in a domain of 5 km in length, 2 km in width, and 1 km in depth, and contains an engineering barrier system with 2,500 deposition holes in fractured crystalline rock.

In this study, a numerical simulation of the reference case is performed with COMSOL Multiphysics as a part of Task F. The fractured crystalline rock is described with the discrete fracture matrix (DFM) model, which expresses major deterministic fractures explicitly in the domain and minor stochastic fractures implicitly with upscaled quantities. As an output of the numerical simulation, fluid flow at steady-state and radionuclide transport are evaluated for $\sim 10^6$ years. The result shows that fractures dominate the transport of radionuclides due to much higher hydraulic properties than rock matrix. The numerical modeling approaches used in this study are expected to provide a basis for performance assessment of nuclear waste disposal repository located in fractured crystalline rock.

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Keywords: DECOVALEX-2023 Task F, APro, Crystalline rock, Fracture, Discrete fracture matrix

Potential of Aqueous U(VI) Removal by Indigenous Bacteria Present in Deep Granitic Groundwater

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Deep geological repository (DGR) has been considered as a globally accepted strategy to dispose high-level radioactive wastes. During long storage periods of 100,000 years, uranium (U) could be migrated through fractures in deep granite aquifers and interact with indigenous bacteria under anaerobic condition. Anaerobic bacteria can reduce U(VI) and further precipitate in the form of U(IV)-oxide minerals by transferring electrons through c-type cytochrome. In this point of view, a comprehensive understanding of uranium-microorganisms interaction is necessary to guarantee the safety of high-level radioactive waste disposal. Although diverse bacterial communities are present in DGR environment, a number of studies have been focused on some model bacteria, such as *Desulfovibrio*, *Geobacter*, and *Shewanella* spp.. In this study, indigenous bacterial community in deep granitic groundwater at 234–244 m was inoculated to sterile uranium-contaminated granitic groundwater amended with 20 mM of sodium acetate, and then incubated under anaerobic condition for 12 weeks. Bio-reduction of U(VI) to U(IV) by indigenous bacteria in uranium-contaminated groundwater was investigated during whole operation period. Initial U(VI) concentration of $885.4 \mu\text{g}\cdot\text{L}^{-1}$ gradually decreased to $586.1 \mu\text{g}\cdot\text{L}^{-1}$, resulting in approximately 33.8% of aqueous U(VI) removal efficiency. Oxidation-reduction potential (ORP) value was gradually decreased from 175.4 mV to -243.0 mV after the incubation of 12 weeks. The decrease in ORP value was attributed to the presence of aerobic bacteria and facultative anaerobic bacteria in indigenous bacterial community. The shift in bacterial community structure was observed by 16S rRNA gene high-throughput sequencing analysis. *Proteobacteria* (55.6%), *Firmicutes* (24.1%), *Actinobacteria* (5.5%), and *Bacteroidetes* (5.4%) were dominant in initial indigenous bacterial community, while *Proteobacteria* (94.8%) was found to be the only abundant phylum after the reaction. In addition, great increase in the relative abundance of sulfate-reducing bacteria (SRB) was observed: the relative abundance of SRB increased from 11.4% to 44.3% after the reaction. This result indicates that the SRB played a key role in the removal of aqueous U(VI). This finding shows the potential of aqueous U(VI) removal by indigenous bacteria in DGR environment.

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Keywords: Deep geological repository (DGR), Uranium, Bio-reduction, Indigenous bacteria, Illumina Miseq analysis, Sulfate-reducing bacteria (SRB)

Investigation of Homogeneity for the Bentonite Blocks

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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 bentonite buffer is one of the most important components to assure the safe disposal of high-level radioactive waste (HLW). As the bentonite buffer is installed as a block type, it is important to fabricate homogeneously. Generally, floating die method and cold isostatic press (CIP) method are used to fabricate bentonite blocks. In this paper, two bentonite blocks were produced using float die method at first, and CIP method was additionally applied to just one block. After that, several samples were cored from two blocks. The dry density and water content of several samples produced from two blocks were measured.

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Keywords: Bentonite block, Homogeneity, Dry density

Some Notes on the KBS-3V Disposal Concepts for Spent Nuclear Fuels in Korea

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Around 40 years ago, in the mid-1980s, Swedish government approved the KBS-3 method for the direct disposal of spent nuclear fuels (SNF) in Sweden. Since then, this method has become a reference for many countries including Korea, Republic of. The main ideas of the KBS-3 method are to locate SNF at 500 m below the ground surface using a copper disposal canister and a bentonite buffer. In 2016, our government announced the National Plan (NP 2016) regarding the final management of high-level waste (HLW) in Korea. In 2019, new committee were organized to review the NP 2016, and they submitted the final recommendations to the government in 2021. Finally, the government announced the 2nd National Plan in December, 2021. So far, KAERI has developed the technologies related to the final management of SNF in two directions. One follows ‘direct disposal’ based on the KBS-3 concept, and the other ‘recycling’ based on ‘pyroprocessing-and-SFR’ (PYRO-SFR).

Even though Posiva and SKB obtained the construction permits with the KBS-3 method in Finland and Sweden, respectively, there are still several technical obstacles to applying directly to our situations. Some examples are as follows: high burnup, huge amounts of SNF, and high geothermal gradient in Korean peninsula. In this work, we try to illustrate some limits of the KBS-3 method. Within our country, currently, the most probable disposal option is the KBS-3 type geological disposal, but no one knows

what the best option will be in 20 or 30 years if those kinds of drawbacks are considered. That is, we compare the effects of the drawbacks using our geological data and characteristics of spent fuels.

Last year, we reviewed alternative disposal concepts focusing on the direct disposal of SNF and compared the pros and cons of them in order to enhance the disposal efficiency. We selected four candidate concepts. They were multi-level disposal, deep borehole disposal, sub-seabed disposal and mined deep borehole matrix.

As mentioned before, KAERI has developed a pyroprocessing technology based on the SFR to reuse fissile radionuclides in SNF. Even though we can consume some fissile nuclides such as ²³⁹Pu and ²⁴¹Pu using PYRO-SFR cycle, there still remain many long-lived radionuclides such as ¹²⁹I and ¹³⁵Cs waiting for the final disposal. The authors review and propose several concepts for the future final management of the long-lived radionuclides.

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Keywords: Spent nuclear fuel, Disposal concepts, KBS-3V, Pyro-processing, Long-lived radionuclide

Investigation of Regulatory Review for Each Development Case of Deep Disposal System in Overseas Leading Countries

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Many countries plan to dispose of spent nuclear fuel through deep geological disposal system. In Korea, a plan is being established for the construction of a deep disposal facility to dispose of high-level radioactive waste (or spent nuclear fuel). For construction of a deep geological repository, the NSSC (Nuclear Safety and Security Commission) stipulate that detailed technical standards for location, structure, and disposal system of deep geological repository are determined and announced by the Nuclear Safety and Security Commission Notification. Therefore, the regulatory body should carry out the process of regulatory review whether the technical standards developed by the implementer are suitable for the IAEA's recommendations and guidelines and domestic conditions. In this process, there are many difficulties and uncertainties in terms of time and cost to independently develop safety factors in Korea by referring to the IAEA reports. So, this study intends to investigate and analyze regulatory cases for important safety factors through cases of overseas leading countries in deep geological disposal project. There are two regulatory cases intensively investigated in this study. The first is a regulatory case of regulatory bodies and external experts on the safety case, and the second is a regulatory review case in the process of site selection factor selection. In case of regulatory review of safety case, Sweden and France were selected as the representative target countries. In Sweden, safety cases such as SR-97, SR-Can, and SR-Site have been developed and there are cases of active regulatory review by regulatory agencies in the RD&D process. In France, several safety cases based on sedimentary rocks were developed and the OECD/NEA IRT (International Review Team) was inquired for review for each safety case. The site selection process is divided into a preliminary site selection stage, a site investigation stage, and a site selection and application stage. In each stage, evaluation to select a safe site is carried out using allocated siting factors of that stage. The IAEA SSG-14 report describes aspects that implementers consider in the site selection process and, with this reference, many countries are developing various siting factors and assessment methodologies in consideration of their domestic bedrock condition and geological positions. As a representative example, in Japan which is highly affected by earthquakes and igneous activities, the siting factor is classified into EF (Evaluation Factors) and FF (Favourable Factors). So, site assessment is conducted preferentially using EF related to earthquakes and igneous activity.

Keywords: Siting factor, Safety case, Deep geological repository, Regulatory review

Corrosion Behavior of 10-year-old Copper Specimens in a 30°C Aerobic KURT Environment

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Corrosion cells that simulates engineering barrier system have been stored in an aerobic KURT environment for 10 years, which were recovered and dismantled in 2021. The test specimens were compressed copper (Com. Cu), Cold spray copper (CSC Cu), Ti Gr.2, STS 304, and Cast nodular iron. The specimens were buffered by compact Ca-type Gyeongju bentonite (KJ-I) and compact Na-type Wyoming bentonite. And the corrosion cells were exposed to KURT groundwater at 30°C for about 10 years (3,675 days). As a result of the long-term experiment in aerobic environment, it was confirmed that Na-bentonite is more advantageous for inhibiting corrosion than Ca-bentonite. The corrosion thickness of the most specimens in Ca bentonite was slightly lower than in Na bentonite until the initial 500 days, but after 10 years, the corrosion thickness of copper and cast iron specimens in Na bentonite was clearly lower. The corrosion thickness of the copper specimen in Na bentonite was very low about 0.5 μm in both Com. Cu and CSC Cu. Moreover, the corrosion thickness in Ca bentonite was very high about 4 μm for Com. Cu and 6 μm for CSC Cu. In the case of cast iron, the corrosion thickness in Na bentonite was about 13 μm , and 15 μm in Ca bentonite. The common feature of copper and cast iron specimens in Ca bentonite, which showed a high corrosion thickness, is the forming of a white mineral deposition layer on the specimen surface, which was presumed to be some kind of feldspar. On the other hand, it was found that the STS304 and Ti specimens were hardly corroded even after 10 years. In conclusion, when a white mineral deposition layer was formed on the specimen surface, the corrosion thickness always increased sharply than before, and thus it was estimated that the generation of the mineral deposition layer cause the increase of bentonite permeability, and rather the weakening of existing passive corrosion film.

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

Conceptual Understanding of the Influences by Climate Change on the Long-Term Evolution of the Surface Environment Components

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It can take hundreds of thousands of years for decreasing radiological effects of high-level radioactive wastes to those of natural background radiation. Therefore, long-term time scale should be considered in order to demonstrate performance and safety of deep geological disposal of the radioactive wastes. The changes of surface environment for these long-term time scale can have influence on safety analysis by changing transport path of radionuclides from the radioactive wastes. Changes in climate is considered as one of main factors causing the long-term changes of the surface environment. The own effects and interactions of climate with other components of the geological disposal system are organized in features, events, and processes (FEPs). In this study, some natural processes occurred by changes of climate were suggested and the connectivity between each process is proposed based on the relation of the FEPs concerned with the changes of climate and surface environment. The processes were classified into global and regional/local scales and was analyzed, respectively. Then, the influences of the processes on shallow groundwater and surface water body environment, which might be transport path of radioactive nuclides in local/site scales, were expected. As the proposed connection demonstrate the order or hierarchical relations of the natural processes, it can shows that some output by a certain process may be input of other process connected the former process in numerical simulations for interpreting the processes. If the connection may be considered to be suitable to represent long-term changes of the surface environment, it can be evaluated that the expected scenarios based on the connection is also proper. In addition, it can be helpful in selecting factors to be studied more detailed in terms of climate change for expecting long-term changes in the surface environment by analysis on the input and output data. The results of this study can be used as basic approaches to represent the long-term changes in the surface environment caused by specific natural processes from changes of climate. It will be also helpful for formulating scenarios related to long-term evolution in the surface environment required for performance and safety assessments of the deep geological disposal.

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Keywords: Long-term evolution, Surface environment, Climate change, Geological disposal, Radioactive wastes, Features, events, and processes (FEPs)

Experimental Study of Nuclide Behavior Acceleration Induced by Gas-mediated Transport

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The radioactive waste disposal systems should consist of engineering and natural barriers that limit the leakage of radionuclide from spent nuclear fuel and fundamentally block groundwater from contact with radioactive waste. These considerations and criteria for designing a disposal system are important factors for the long-term stability evaluation of deep geological repository. Colloids and gases that may occur in the near-field and groundwater infiltrated from outside can be means to accelerate the behavior of radionuclide. The gas produced and infiltrated in the disposal system is highly mobile in the porous medium, and reactive gases in particular can affect the phase and behavior of radionuclide. A free gas phase (bubble) can be formed inside the canister if the partial pressure of the generated gas exceeds the hydrostatic pressure. If the gas pressure exceeds the critical endurance pressure of canister and buffer, then a gas bubble may push through the canister perforation and the buffer. It is also known that when gas bubbles are formed, radionuclide or colloids are adsorbed on the surface of the bubbles to enable accelerated movement. An experimental setup was designed to study the acceleration of nuclide behavior induced by gas-mediated transport. A high temperature and pressure reaction system that can simulate the deep disposal environment (500 m underground) was designed. It is also designed to install specimens to simulate gas flow in engineered barriers and natural barriers. The experimental scenario was set based on 1,000 years after the closure of the repository. According to the previous modeling results, the surface temperature of the canister is about 30 to 40 degrees and the gas pressure can be generated between the canister and the buffer is 5 MPa or more. In the experimental conditions, the saturation time of compacted bentonite was measured and the gas permeability of the compacted bentonite according to the dry density was also measured. Further studies are needed on the diffusion of dissolved gas into the compacted bentonite and the permeation phenomenon due to gas overpressure.

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Keywords: Nuclide behavior, Acceleration, Bentonite, Gas bubble, Underground repository system, Canister corrosion

Infrared and Raman Spectra of Domestic KJ-II and MX-80 Bentonites

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FTIR (Fourier Transform Infrared) and Raman spectra of KJ-II bentonite provided by Clariant Korea were compared with those of MX-80 bentonite. The FTIR spectra were obtained using a Nicolet 5 FTIR spectrometer (Fisher Scientific) equipped with a diamond ATR (Attenuated Total Reflection) module. The spectra were collected for 64 scans with a resolution of 4 cm^{-1} . Raman spectra were obtained using an optical microscope (Olympus, BX43) and a spectrometer (Andor, SR-500). The laser beam was focused using an objective lens with a magnifying power of 50. The backscattered light from the sample was collected into an optical fiber with a core diameter of 0.4 mm. The Raman signals were recorded with CCDs (Andor, DV401A-BV for 532 nm laser wavelength and DV420A-OE for 638 and 785 nm laser wavelengths). Each pixel of CCD received the signal for 1 s and its 1000 times accumulated data were collected.

The FTIR spectra of the two bentonite samples are very similar. The FTIR spectra of both bentonites showed absorption bands at $3623, 3399, 3231\text{ cm}^{-1}$ in the higher wavenumber region and at $1637, 1443, 1117, 997, 914, 887, 847, 797, 611, 515, 414\text{ cm}^{-1}$ in the lower wavenumber region. A sharp band at 3623 cm^{-1} and the strong band at 997 cm^{-1} correspond to the OH stretching of structural hydroxyl groups and the Si-O stretching vibration, respectively. In addition to these clear bands, several absorption bands observed in this experiment are well matched with the results reported in various literatures.

Unlike the FTIR spectrum, it is not easy to observe the Raman bands of bentonite. The reason is that strong fluorescence interferes with the Raman spectrum. The two bentonite samples showed different fluorescence intensities. In the case of MX-80 bentonite, no clear Raman band was observed due to the influence of very strong fluorescence. KJ-II bentonite showed a relatively weak fluorescence intensity and Raman bands were partially visible at around $147, 260, 397, 709, \text{ and } 1279\text{ cm}^{-1}$. In particular, the band at 1279 cm^{-1} is wide and sturdy. It was observed that the non-powder samples showed a better quality spectra. The Raman characteristics of KJ-II bentonite, which depend on the incident laser wavelength and the sample pretreatment, are discussed in detail.

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Keywords: FTIR, Raman, Bentonite, KJ-II, MX-80

Identification of Mechanical Parameters of Korean Domestic Bentonite With Machine Learning Technique

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Various models have been proposed to describe the swelling behavior of buffer in high level waster repository. One of the most notable models, the Barcelona Basic Model (BBM), is a mechanical model that simulates the behavior of unsaturated ground and is widely applied to soils that undergo large expansion due to water. Among the BBM parameters of Kyeongju bentonite, which is found in Korea, there are no experimental data for parameters that describe the unsaturated state. Such hydro-mechanical properties should be characterized through experimental programs. However, such experiments are highly complicated and require long periods of time to produce an unsaturated state through different methods according to the suction range. Although there are several studies in which geotechnical parameters were obtained through a back analysis instead of direct experiments, few studies have employed machine learning methods for the identification of geotechnical parameters. In this study, instead of direct experiments, the results of a relatively simple swelling pressure experiment was compared to the numerical analysis results to propose a method of determining some of BBM parameters. Influential factors were identified by a sensitivity analysis and the values of the factors were estimated using an artificial neural network and optimization method. The obtained parameters were applied to the numerical model to estimate the swelling pressure growth, which was subsequently compared to the experimental value. As a result, it was found that there was no significant difference between the two swelling values.

Keywords: Bentonite, BBM, Machine learning, Optimization

Evaluating Leaching Property of Gadolinium-Iron-Garnet Ceramic Waste Form

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Garnet is one of the promising ceramic waste forms for immobilizing radioactive wastes. It has an $A_3^{[VIII]}B_2^{[VI]}T_3^{[IV]}O_{12}$ structure, so it can accommodate various cations of different sizes and coordination. Silicon usually occupies the centers of the tetrahedron structural site ($T^{[IV]}O_4$) in natural garnet. However, substitution of the T-site with iron, which has a relatively large ionic radius, causes the expansion of a unit cell volume of garnet and allows the incorporation of large cations such as actinides at other sites. Relatively few leaching data have been reported for ferrite garnet waste forms to date.

In this study, we synthesized gadolinium-iron-garnet and evaluated the leaching property using cerium as a surrogate for actinide elements. The test specimens were made by cold pressing and sintering process. Three different standard leaching tests were performed as follows. The PCT-A (ASTM C1285) was performed for 7 days at 90°C to the crushed sample (0.149 to 0.074 mm). The ANSI/ANS-16.1 standard leach test was performed at ambient conditions for 5 days with constant replacement of leachate. Finally, the MCC-1 (ASTM C1220) test was performed for 28 days at 90°C with different types of leachants such as ultrapure water, brine, and silicate water. The last two leaching tests were conducted on monolithic specimens. After the end of the test, leachate was analyzed by inductively coupled plasma mass spectroscopy (Agilent, ICP-MS 7700S).

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Keywords: Garnet, Ceramic waste form, Leaching test, PCT, ANSI/ANS-16.1, MCC-1

Dual-Permeability and Dual-Porosity Model for Estimating Canister Corrosion Depth and Canister Lifetime

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To dispose of spent nuclear fuel, the most promising method is disposal in a deep geological repository with a multi-barrier system. Among the multi-barrier system, canisters are used to contain the spent nuclear fuel. A role of the canister is to withstand corrosion load from the deep geological environment as possible as long. Corrosion processes consist of corroding agents transport to the canister surface and electrochemical reactions between the corroding agents and the canister surface. According to previous King's electrochemical experiments, the mass-transport rate of corroding agents is slower than the electrochemical reaction rate with copper when the canister is surrounded by dense bentonite blocks. Therefore, the mass-transport rate is a rate-determining step for the whole corrosion process. Despite of the importance of transportation of oxidizing agents in bentonite, the transportation process was not paid attention. For example, existing models which are called continuum models assumed that the corroding agents pass through the pore in the porous medium because the continuum model does not consider the fracture networks in the bentonite. Here we develop a dual-permeability and dual-porosity model. In this model, the transport of corroding agents is considered that they pass through fracture within the porous medium. The difference between the dual-permeability and dual-porosity model is whether the corroding agents can pass through the pore. The dual-permeability model assumed that the mass-transport occurs within both fracture and porous medium. On the other hand, the dual-porosity model assumed that the mass-transport occurs only within fractures. Through both models, we found that the transport rate in the fractures is much higher than through the pores, and the canister lifetime at a point where contacting the fracture tip is much shorter than other parts when the canister lifetime is calculated by the transport-governed condition. In addition, the temperature distributions in the fracture are different compared to the existing continuum model. Our results show the effect of fractures in terms of not only corroding agents transport but also the canister lifetime. We anticipate our model to be a first step for the corrosion estimation model coupled with fracture networks.

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Keywords: Dual-permeability model, Dual-porosity model, Transport-governed condition, Canister lifetime estimation, Deep geological repository

Effects of EDTA on Leaching Behaviors of Co(II) and Cs(I) Immobilized in Cement Specimen

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Numerous low-and intermediate level radioactive wastes were generated from the decommissioning processes of nuclear power plants. Radionuclides such as Co and Cs contained in decommissioning wastes should be immobilized to prevent the release of radionuclides from the wastes due to its harmful impacts on ecosystem by high radioactivity and long half-life. Ethylenediaminetetraacetic acid (EDTA) used as decontamination agent can be contained in cement waste during decommissioning process of nuclear power plants. In addition, EDTA can be stably and strongly bound with radionuclides, resulting in the acceleration of the nuclide release from solidified cement matrix. Here, we investigated the effects of EDTA on leaching behaviors of Co and Cs immobilized in the cement specimen. The leaching tests were performed according to the ANS 16.1 “Measurement of the leachability of solidified low-level radioactive wastes by a short-term test procedure”. From the results, an increase in the EDTA content in the cement specimen led to an increase in Co leaching, whereas a decrease in Cs leaching. Leaching of Cs was dominantly controlled by diffusion from the pore space of the cement specimen to the solution. The effective diffusion coefficient and leachability index of nuclide were determined using the diffusion-release models of ANS 16.1. The results of present study can be used in the safety assessment for disposal of the radioactive waste generated by decommissioning of nuclear power plants.

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Keywords: Cobalt, Cesium, Leaching behavior, EDTA, Cement, Decommission

Grade Improvement of Montmorillonite in Bentonite Using Selective Separation Method

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Bentonite is considered as buffer of engineered barrier for retardation of radionuclide migration. Bentonite has low permeability, high swelling and high sorption capacity for radioactive nuclides. Properties have been widely investigated under various geochemical conditions simulating deep geological environments. The chemical stability of bentonite is an important factor in evaluating the long-term stability of the bentonite buffer. However, the presence of impurities in bentonite clays can reduce the retention capacity for retardation of radionuclide migration value of bentonite. Therefore, the bentonite purification is necessary. In the present study, grade improvement of montmorillonite was conducted using ultrasonic and froth flotation methods. As a result of confirming the grade of montmorillonite according to the optimal ultrasonic intensity for ultrasonic irradiation is 1.0 kHz of bentonite in Gyeongju (KJ-II) increased from 60% to 78%. In case of froth flotation method using PSS (0.1 mM) as a reagent, the grade of montmorillonite increased up to 90%.

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Keywords: Montmorillonite, Bentonite, Grade improvement, Ultrasonic, Froth flotation

Development of Numerical Model for the Long-term Evolution due to Mechanical Behavior Considering Excavation Damaged Zone

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In recent years, the importance of the thermo-hydraulic-mechanical-chemical coupled processes is increasing in the performance assessment (PA) of the high-level radioactive waste repository. In the case of mechanical behavior, it is very important because it can affect fluid flow and radionuclide transport by changing the porosity and permeability of the medium. In particular, Excavation Damaged Zone (EDZ) should be considered essential in PA because the migration of radionuclide is affected by the enhanced hydraulic transmissivity and altered geomechanical behavior of EDZ. Furthermore, due to various thermo-hydraulic behaviors such as decay heat generated from radioactive waste, pore water pressure increase, and swelling pressure of bentonite buffer material, mechanical evolution is occurred which may change the size and physical properties of EDZ. Therefore, to solve this problem, analysis of coupled thermal-hydraulic-mechanical (THM) processes with the effect of long-term evolution of EDZ due to the mechanical behavior should be accompanied. In this study, numerical model for the long-term evolution due to mechanical behavior considering EDZ using the Adaptive Process-based total system performance analysis framework for a geological disposal system (APro) proposed by the Korea Atomic Energy Research Institute (KAERI). In the case of EDZ, the concept of Mazars' damage evolution model was applied to simulate the behavior using the continuum model, and the change in hydraulic properties according to the degree of damage was considered. To investigate the importance of mechanical behavior in PA, the results were compared by performing numerical analysis according to the presence or absence of mechanical analysis. Finally, numerical analysis considering the mechanical evolution of EDZ was conducted using the model developed in this study to investigate the effect of EDZ.

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Keywords: High-level radioactive waste repository, Process-based total system performance assessment, APro, Excavation damaged zone, Long-term mechanical evolution

Geochemical Characteristics of Granite From KURT for Sorption and Diffusion Experiments

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The deep geological repository consisting of a multi-barrier system (engineered and natural barriers) is generally designed to isolate the high-level radioactive waste. The natural barrier is outermost portion to secure safety of the disposal. Crystalline rocks are considered for potential geological repository media to retard and inhibit the migration of radionuclides when the radionuclides leak from the canister and break through the engineered barrier. Sorption and diffusion processes play a major role in retardation of the radionuclides in deep underground environment. In order to evaluate the migration of radionuclides in the safety assessment or geochemical modelling, distribution coefficient and diffusivity of radionuclides are required as input data. In this study, we performed mineralogical and geochemical analysis for a crystalline rock (e.g., granite) to use the sorption and diffusion experiment. The fresh rock samples are obtained from a deep core samples (DB-2) drilled up to 1 km from the surface at KURT (KAERI Underground Research Tunnel) site. For the optical and microscopic examination, thin sections of the rock sample were provided. The rock samples were crushed into powder size to analyze major and trace elements of the whole-rock aliquots. The powdered specimens also used for mineral identification and measurement of specific surface area. The major constituent minerals of the granite are plagioclase, quartz, and K-feldspar and the minor minerals are phlogopite, biotite, and chlorite. According to the results of geochemical analysis, the granite specimens generally contain more than 70wt% of SiO_2 and 8wt% of total alkali oxides ($\text{Na}_2\text{O} + \text{K}_2\text{O}$). The trace elements normalized to primitive mantle compositions show positive Cs, Rb, U, K, and Pb anomalies and negative Nb and Ti anomalies. The rock samples have an average density of $2.62 \text{ g}\cdot\text{cm}^{-3}$ and an average porosity of 0.222%. The crushed samples represent the specific surface area of $0.2087 \text{ m}^2\cdot\text{g}^{-1}$ for the 75–150 μm fraction and $0.1616 \text{ m}^2\cdot\text{g}^{-1}$ for the 150–300 μm fraction by BET method, respectively. The granite specimens will be used for the sorption and diffusion experiments to evaluate the radionuclides' geochemical behaviors. The mineralogical and geochemical properties provided in this study can be useful in understanding the sorption and diffusion processes of significant radionuclides under the geological disposal environments.

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Keywords: Crystalline rock, Granite, Geochemistry, Sorption, Diffusion, Deep geological disposal, Natural barrier

Analysis of the Effect of the Heat on Groundwater Flow in the Fractured Crystalline Rock Near the HLW Repository

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The design of the high-level radioactive waste (HLW) repository is made for isolating the HLW from the groundwater system by using artificial and natural barriers. Granite is usually considered to be a great natural barrier for the HLW repository in various countries including Sweden, Canada, and Korea due to its low hydraulic permeability. However, many fractures that can act as conduits for groundwater and radionuclides exist in granite. Furthermore, the decay heat generated by the HLW can induce groundwater acceleration through the fracture. Since the direction, magnitude, and lasting time of the heat-induced groundwater flow can be differed depending on the fracture geometry, the effect of fracture geometry on the groundwater flow around the repository should be carefully analyzed. In this study, groundwater models were conducted with various fracture geometries to quantify the effect of various properties of fractures (or fracture networks) on the heat-induced groundwater flow. In all models, the pressure around the repository only lasted for a short period after it peaked at 0.1 years. In contrast, the temperature lasted for 10,000 years after the disposal inducing the convective groundwater flow. Single fracture models with different orientations were conducted to evaluate the variations in groundwater velocities around the repository depending on the fracture slope. According to the results, the groundwater velocity on the fracture was the fastest when the regional groundwater flow direction and the fracture direction coincided. In double fracture models, various inclined fractures were added to the horizontal fracture. Due to the intersecting, the groundwater flow velocity showed a discontinuous change at the intersecting point. Lastly, the discrete fracture network models were conducted with different fracture densities, length distributions, and orientations. According to the modeling results, the groundwater flow was significantly accelerated when the fracture network density increased, or the average fracture length increased. However, the effect of the fracture orientation was not significant compared to the other two network properties.

Keywords: High-level radioactive waste repository, Groundwater, Decay heat, Discrete fracture network (DFN)

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

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The timescale for the post-closure safety assessment of a deep geological repository ranges from ten thousand to a million year. In such a long period of time, the biosphere inevitably undergoes changes. Therefore, the long-term evolution of a biosphere is recognized as an important issue in the post-closure safety assessment of a deep geological repository for spent fuels. In this study, we reviewed the approaches to address the long-term evolution of a biosphere. The major drivers of long-term evolution of a biosphere are the climate change and the resulting landscape development. They can affect the hydrogeological and hydrogeochemical characteristics of a biosphere, and then the radionuclide migration through the biosphere followed by the exposure doses for the critical groups. In addition, human activities and the social developments can affect the climate change resulting in the long-term evolution of a biosphere. To make a biosphere assessment, the long-term evolution scenarios for the biosphere should be formulated considering these climate change, landscape development, and human activities. In addition, features, events, and processes (FEPs) that affect the long-term evolution of a biosphere should be used. According to the Safety Case reports of Finland, the major long-term evolution scenario drivers of a biosphere are local sea-level change due to climate change and land use related to crop type, irrigation procedures, livestock, forest management, construction of a well, and demographics. The climate change causing the local sea-level change can be simulated using various earth system models such as CLIMBER-2, MPI/UW, and UVic and an ice-sheet model such as SICOPOLIS. The review results of this study and FEPs related to the climate change, the landscape development, and human activities will be used to formulate long-term evolution scenarios for the safety assessment of a deep geological repository for spent fuels.

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Keywords: Safety assessment, Deep geological repository, Biosphere, Long-term evolution, Climate change

Verification of Improvement in the Parallel Computation Performance of the Operator Splitting Method in APro

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APro, developed by KAERI as a process-based total system performance assessment model, can simulate the radionuclide transport affected by thermal, hydraulic, mechanical and geochemical changes that may occurs in the engineering and natural barriers of a geological disposal system. APro targets a large-scale and heterogeneous 3D system that includes more than 10,000 boreholes located about 500 m underground and hundreds of fractures of different sizes distributed within an area of several km². Simulating transport and reaction phenomena for such a system through the global implicit approach (GIA) may require considerable computational resources or be intractable in some cases. Therefore, APro adopts the sequential non-iterative approach (SNIA), one of the operator splitting (OS) methods, to separate the mass transport and reaction phenomena into independent problems. By using SNIA, the parallel computation performance in APro with multiple cores is expected to be improved. In this study, the effect of SNIA on the parallel computation performance was analyzed through a simple 1D reactive transport problem. Without SNIA, finite difference equations, discretized from the partial differential equations (PDEs) describing the reactive transport problem, have to be solved at once because all dependent variables are nonlinearly and spatially interconnected through reaction and mass transport terms. When the reaction and mass transport terms are separated through SNIA, the mass transport problem can be converted into independent linear equations for each chemical and the efficient linear system solver can be applied to each linear equation. In particular, since the reaction problem is changed to independent nonlinear equations for each node, the parallel computation performance can be greatly improved. To verify this, the 1D reactive transport problem was implemented in MATLAB, and SNIA and GIA were applied to solve the problem. As a result, there was no significant difference in results between SNIA and GIA for proper spatial and temporal discretization, which verified the accuracy of SNIA. In order to see the parallel computation performance, the calculation times for SNIA and GIA with increasing number of cores were measured and compared. As the number of cores increased, the SNIA calculation speed became faster than that of GIA, which verified that SNIA could improve parallel computation performance in APro. In the future, the effect of SNIA on the parallel computation performance will be verified for the numerical analysis of large-scale geological disposal systems.

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

Analysis of Changes in Reference Spent Nuclear Fuels Decay Heat Applied to the Reference Disposal Concepts

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The research for the safe management of high-level waste in Korea has been conducted by the Korea Atomic Energy Research Institute since 1997, and the results have formed the basis of the national basic plan for the high-level waste management and the revised national basic plan. In the future, it is evolving and developing R&D focusing on securing technologies for demonstration of the disposal technologies and R&D to develop disposal concepts that increase safety and improve efficiency. Efficient management of heat generated from high-level radioactive waste, including spent nuclear fuel, is an important factor in establishing the disposal concepts because it must be in harmony with key factors such as repository layout, waste disposal container specifications, and design and operation for the barriers of the disposal system. For safe and complete isolation of high-level radioactive waste in the deep geology, the disposal systems that meet the thermal requirements for the disposal system design have been developed by harmonizing the thermal characteristics of engineered and natural barriers in Korea. These disposal systems were based on low burn-up spent nuclear fuel characteristics generated in the early stages of nuclear power generation, and next, based on the high-level wastes from recycling process of the high burn-up spent nuclear fuels, and were the direct disposal systems for the high burn-up spent nuclear fuels. So, it is necessary to track and analyze the change process in the decay heat characteristics of the high-level waste to be disposed of in order to improve the disposal concept, which enhances the safety of disposal and the utilization of the national land. Therefore, in this paper, the process of change in decay heat of reference spent nuclear fuels for disposal applied to the disposal concepts from the initial stage of development of high-level waste disposal technology to the present in Korea is analyzed.

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Keywords: High-level waste, Spent nuclear fuels, Decay heat, Deep geological disposal, Thermal management

Examination of the Domain Decomposition Solvers in COMSOL for a Large-scale Problem of APro in Parallel Computing Environment

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Domain decomposition method (DDM) has been widely employed for the numerical analysis of large-scale problems due to its applicability to parallel computing. DDM divides the modeling domain into a set of subdomains and obtains the entire solution iteratively until the values of each subdomain which are shared with other subdomains, such as boundary values, are converged. Therefore, in general, DDM is a memory-efficient iterative algorithm with inherent parallelism on the geometric level. APro, the process-based total system performance assessment model, aims for simulating the radionuclide transport considering coupled multi-physics phenomena occurring in large-scale geological disposal system, which are inevitably accompanied by huge memory burden. Therefore, DDM is applicable for the large-scale problem of APro and its performance in parallel computing needs to be examined.

The DDM solvers provided by COMSOL which constitute APro can be classified into two methods. One is the overlapping Schwarz method that each subdomain overlaps its neighboring domains and the other is the Schur complement method that subdomains are non-overlapping and separated by boundary domains. For the Schwarz method, the additive, hybrid, multiplicative and symmetric methods can be selected according to the solution update scheme. And for the Schur method, the additive and multiplicative ordering options can be chosen for solving Schur complement system.

In this study, the calculation efficiency of the DDM solvers in COMSOL and the applicability to the cluster environment were examined. In aspect of efficiency, the memory requirements with different number of subdomains and calculation schemes were compared in a single node. Then, the memory requirements with increasing number of disposal tunnels and deposition holes were investigated in multiple nodes. As a result, on the cluster environment, with the help of distributed memory architecture which enables efficient memory usage, the applicability of DDM solvers to the large-scale problem of APro was confirmed.

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Keywords: APro, Total system performance assessment, COMSOL, Domain decomposition method, Parallel computing

Feasibility Study on Adopting Hybrid Neural Network Solvers in APro

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APro, a modularized framework of the process-based total system performance assessment, has been developed by KAERI to simulate the radionuclide transport in geological disposal system considering multi-physics phenomena. However, the target problem including more than 10,000 boreholes and over 100,000 years of simulation time is computationally challenging to deal with numerical solvers provided by COMSOL Multiphysics constituting APro. To alleviate the computational burden, machine learning (ML) techniques have been studied to develop a surrogate model replacing the heavy computation part.

In recent studies, attempts have been made to integrate the knowledge of physics and numerical methods into the ML model for partial differential equations (PDEs). Unlike conventional ML approaches solely relying on data-driven method, the integration can help to make the ML model more specialized for solving PDEs. The hybrid neural network (NN) solver method is one of the strategies to develop more efficient PDE solver by interleaving NN with numerical solvers like finite element method (FEM). The hybrid NN model on the premise of numerical solver is easier to train and more stable than the purely data-driven model. For example, one previous study has used the hybrid NN model as a corrector for an incomplete numerical solver for the advection-diffusion problem. In every time step of simulation, NN corrects the error of incomplete solution obtained by a relaxed numerical solver with coarse meshing. The simulation in the next time step starts from the corrected solution, so NN interacts with the numerical solver iteratively. If the corrector is successfully trained, the incomplete but fast solver with corrector can provide reliable results comparable to the original massive solver.

This study adopts the hybrid concept to develop a surrogate model for the near-field region, which is the heavy computation part in the simulation of geological disposal system. Various incomplete models such as coarse meshing or emptying the borehole domain are studied to construct a hybrid NN solver. This study also covers how to embed the hybrid NN in COMSOL Multiphysics to train and use it during the simulation.

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Keywords: APro, Neural network, Machine learning, COMSOL Multiphysics

Construction Plan of Demonstration Test Facilities for An Emplacement and Retrieval Device Applicable Deep Borehole Disposal

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Deep geological disposal (DGD) of spent nuclear fuels (SNF) at 500 m⁻¹ km depth has been the mainly researched as SNF disposal method, but with the recent drilling technology development, interest in deep borehole disposal (DBD) at 5 km depth is increasing. In DBD, up to 40SNF canisters are disposed of in a borehole with a diameter of about 50 cm, and SNF is disposed of at a depth of 2–5 km underground. DBD has the advantage of minimizing the disposal area and safely isolating high-level waste from the ecosystem. Recently, due to an increasing necessity of developing an efficient alternative disposal system compared to DGD domestically, technological development for DBD has begun. In this paper, the research status of canister operation technology and plans for DBD demonstration tests, which subjects are being studied in the project of developing a safety-enhancing high-efficiency disposal system, are introduced.

The canister operation technology for DBD can be divided into connection device development and operation technology. The developing connection device, emplacing and retrieving canisters in borehole, adopted the concept of a wedge thus making replacement equipment at the surface unnecessary. The new connection device has the advantage of being well applied with emplacement facilities only by simple mechanical operation.

The technology of operating a connection device in DBD can be divided into drill pipe, coiled tubing, free-drop, and wireline. The drill pipe is a proven method in the oil industry, but requiring huge surface equipment. The coiled tubing method uses a flexible tube and shares disadvantages as the drill pipe. The free-drop is a convenient method of dropping canister into a borehole, but has a weakness in irretrievability in an accident. Finally, the wireline method can be operational on a small scale using hydraulic cranes, but the number of operated canisters at once is limited.

The test facility through which the connection device is to be tested consists of dummy canister, borehole, lifting part, monitoring part, and connecting device. The canister weight is determined according to the emplacement operation unit. The lifting part will be composed following wireline consisting of a crane, a wire and a winding system. The monitoring part will consist of an external monitoring system for hoists and trolleys, and an internal monitoring system for the connection device's location, pressure, and speed. In this project, a demonstration test will be conducted in a borehole with 1km depth, 10 cm diameter provided by KAERI to verify operation in the actual drilling environment after design improvement of the connecting device. If a problem is found through the demonstration test, the problem will be improved, and an improved connection device will be tested to an extended borehole with a 2 km depth, 40 cm diameter.

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Keywords: Deep borehole disposal, Spent nuclear fuel, Disposal canister, Connection device, Wireline

Investigation of Operating Limit for Accelerated Life Test of Monitoring Sensor

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The high-level nuclear waste disposal system is a structure with a very long life expectancy, and deterioration and cracking of the structure may occur over time. In addition, the high-level nuclear waste disposal system is in complex extreme conditions such as high temperature, groundwater, and radiation. Therefore, we need to develop a highly durable monitoring sensor that can detect the deterioration and crack of structures in extreme conditions. Since the durability of a sensor is closely related to the sensor lifetime, it is essential to predict the sensor lifetime accurately. The sensor lifetime can be predicted through the reliability qualification test. Among them, the accelerated life test conducted under harsh conditions is widely used as a method to shorten the test period. The major factor in carrying out the accelerated life test is to set the appropriate harsh conditions. Therefore, this study experimentally derived the operating limit of the monitoring sensor. It is essential to set the proper harsh conditions when performing the accelerated life test. Through this study, it is judged that it will be helpful in determining the appropriate stress level when performing the accelerated life test for accurate lifetime prediction.

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Keywords: High-level nuclear waste, Monitoring sensor, Operating limit, Accelerated life test, Reliability qualification test

Swelling Indices of Bentonite-Sand Mixtures According to Sand Addition Rates

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Increasing thermal conductivity of buffer materials makes the disposal tunnel and hole spacings in high-level radioactive waste (HLW) repositories decrease, so that the area of the HLW repository decreases, which gives more choices to choose the HLW repository site and economical cost savings to construct HLW repositories. Thus, developing enhanced buffer materials with improved thermal conductivity is needed. One of the methods to develop enhanced buffer materials is to add additives to the bentonite which is main material for buffer materials. Most additives have high thermal conductivity, but most additives do not swell or less swell than bentonite, so that the swelling pressure of the enhanced buffer materials by additives decreases compared to the swelling pressure of pure bentonite buffer materials. Swelling pressure is an important performance criterion to design buffer materials. Thus, it is important to confirm the swelling pressure of the enhanced bentonite. However, it is not simple to measure the swelling pressure of the buffer materials and furthermore, it takes several days to measure the swelling pressure of the buffer materials. For these reasons, swelling index can be considered to predict the swelling pressure of the enhanced buffer materials relatively. In this study, it was investigated through tests how the swelling index of bentonite-sand mixtures change according to the amount of sand and it was found that the linear relationship between swelling index and sand amount in the bentonite-sand mixtures.

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Keywords: High-level radioactive waste repository, Enhanced buffer material, Swelling index, Additive

A Study on the Evaluation of Backfill Candidate as to Performance Requirement in Deep Geological Disposal

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The backfill close the deep geological disposal system by filling the disposal tunnel and the connecting tunnel after the installation of buffer in the disposal hole. SKB and Posiva have established and designed the safety function of the backfill for the common goal of the deep geological disposal system. The safety function of backfill material has been set hydraulic conductivity of less than $10^{-10} \text{ m}\cdot\text{s}^{-1}$, a swelling pressure of 0.2 MPa, a compressive modulus of 10 MPa or a buffer density of $1,950 \text{ kg}\cdot\text{m}^{-3}$ or more, and freezing resistance. For the selection of the optimum backfill material, SKB and Posiva developed the concept of the backfill and evaluated the candidate that satisfies the requirements in four steps. In the first step, the performance and function that the backfill material should have were conceptualized. For the second step, laboratory tests and in-depth analysis of the candidate material properties were conducted. At this step, the focus has been on testing with the concept of the block method, using key candidate materials. In step 3, laboratory and large-scale experiments were performed to test engineering feasibility. In addition, design specifications for backfill materials were set based on site conditions, installation methods, and short- and long-term functions of materials. In Korea, it is only now in the step of selecting the concepts of the safety function. Therefore, it is necessary to benchmark the development process based on the previous studies of SKB and Posiva. In this study, candidate materials, experimental methods, and results were analyzed. As a result, the research steps and conditions for the selection of the optimum backfill material were reviewed. Using this study, the research steps of domestic backfill was suggested to develop within a short time for the Korean deep geological disposal system.

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Keywords: Disposal repository, Backfill, Candidate material, Performance requirement

Distributions of Thermal Performance and Deposition Hole Spacing of High-level Radioactive Waste Repository Based on Rock Thermal Properties and Geothermal Gradient in Korea

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To decrease area of the repository for high-level radioactive waste, enhancing the disposal efficiency is needed for public acceptance. Previous studies regarding the performance assessment of KRS and KRS+ repository did not consider area-based variations of the geothermal gradient and rock thermal properties in Korea. This research estimated deposition hole spacing based on performance assessment of a repository using the distribution of geothermal gradient and rock thermal properties in Korea to increase disposal efficiency. Distributions of geothermal gradient, rock thermal properties were investigated based on 2019 Korea geothermal atlas published by Korea Institute of Geoscience and Mineral Resources (KIGAM). Effect of thermal performance parameters was analyzed using coupled thermal-hydraulic numerical simulations, and effect of rock thermal conductivity and deposition hole spacing on the maximum temperature of buffer was relatively large. In addition, distribution maps of thermal performance of a repository and deposition hole spacing were plotted using thermal performance parameters-maximum temperature of buffer regression equations and GIS data given by KIGAM. In the regions showing the highest maximum temperature of buffer in Korea, required deposition hole spacings were 10.5 m, 10.0 m, 10.1 m, respectively for KJ-II, MX-80, and FEBEX bentonite cases, and thereby additional disposal area of 40%, 33.3%, and 34.7% were required compared to that of the KRS+ repository. On the other hand, high disposal efficiency can be obtained in the regions showing the low maximum temperature of bentonite buffer. The methodology provided in this research can be used as one of the references for the selection of domestic candidate repository sites. Additional mechanical performance analysis should be conducted using distributions of mechanical properties of rock mass in Korea.

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Keywords: High-level radioactive waste disposal, Thermal-hydraulic numerical simulation, Geothermal gradient and rock thermal properties in Korea, Deposition hole spacing

Criticality Analyses of Two Different Disposal Canisters for Deep Geological Repository in Burnup Credit

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The criticality analyses considering burnup credit were performed for a spent nuclear fuel (SNF) disposal cell consisting of bentonite buffer and two different types of PWR SNF disposal canister: the KBS-3 type canister and the small standardized transportation, aging and disposal (STAD) canister. The criticality analyses were carried out for four cases as follows: (1) the calculation of isotopic compositions within a SNF using a depletion assessment code and (2) the calculation of the effective multiplication factor (k_{eff}) value using a criticality assessment code. Firstly, the KBS-3 type canister containing four SNFs of the initial enrichment of 4.0wt% ^{235}U and discharge burnup of 45,000 MWD/MTU was modelled. The k_{eff} values for the cooling times of 40, 50, and 60 years of SNFs were calculated to be 0.74407, 0.74102, and 0.73783, respectively. Secondly, the STAD canister was modelled. The SNFs contained in the STAD canister were assumed to be the enrichment of 4.0wt% and the burnup of 45,000 MWD/MTU. The k_{eff} values for the cooling times of 40, 50, and 60 years were estimated to be 0.71448, 0.70982, and 0.70743, respectively. Thirdly, the KBS-3 canister with four SNFs of which the enrichment was 4.5wt% and the burnup was 55,000 MWD/MTU was modelled. The k_{eff} values for the cooling times of 40, 50, and 60 years were 0.73366, 0.72880, and 0.72634, respectively. Finally, the calculations were carried out for the STAD canister containing four SNFs of the enrichment of 4.5wt% and the burnup of 55,000 MWD/MTU. The k_{eff} values for the cooling times of 40, 50, and 60 years were 0.70323, 0.69946, and 0.69719, respectively. Therefore, all of four cases met the performance target with respect to the k_{eff} values, 0.95. The STAD canister showed lower k_{eff} values than the KBS-3 canister. This appears to be the neutron absorber plate installed in the STAD canister although the distance among the four SNFs in the STAD canister was shorter than the KBS-3 canister.

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Keywords: Criticality analysis, Burnup credit, Deep geological repository, Disposal canister, Effective multiplication factor

Iodide Crystallization Using a Copper Carbonate Material

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A method to effectively scavenge highly mobile radioiodide into a solid material was developed. Under an anaerobic condition, as copper(II) was strongly associated with bicarbonate (HCO_3^-) in solution, malachite quickly formed, and then it was gradually transformed to a compact crystal of CuI (marshite) attracting iodide. The formation of CuI crystal was principally led by the spontaneous Cu-I redox reaction centering around the copper phase over the presence of sulfate (SO_4^{2-}). The transformed CuI crystal was poorly soluble in water. Interestingly, this redox-induced iodide crystallization was rather promoted over the existence of anionic competitors (e.g., HCO_3^- and SO_4^{2-}). Unlike the conventional methods, these competing anions positively behaved in our system by supporting that the initial malachite was more apt to be reactive to largely attract highly mobile I^- . Under practical environments, such a selective I^- uptake and fixation into a crystalline form will be a promising way to effectively remove I^- in a great capacity.

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Keywords: Iodide, Malachite, Marshite, CuI, Iodide crystallization

Review on Screening Process of Radionuclides List for a Safety Assessment of Geological Disposal

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An objective of a safety assessment for geological disposal is to evaluate the radiological impact by radionuclides release from radioactive wastes. Computational estimation of all radionuclides transport in the disposal system, however, is not necessary because some radionuclides has negligible effect on radiological doses. For this reason, prioritization of radionuclides list is preceded before the safety assessment. The Korea Atomic Energy Research Institute (KAERI) has assessed the long-term safety of a disposal system for spent nuclear fuels. Currently, thirty eight radionuclides and twenty three elements are considered in the safety assessment activity of the KAERI. Nevertheless, a screening process for radionuclides selection has not been articulated yet.

In this study, we reviewed radionuclides selection process in foreign countries to re-establish screening criteria for the KAERI's radionuclides list. Screening models of the Swedish Nuclear Fuel and Waste Management Company (SKB), the Department of Energy (US DOE), and the Japan Nuclear Cycle Development Institute (JNC) were compared. We found that each country developed different screening model depending on scenarios of radionuclides release. Nonetheless, there were common properties that determines the importance of radionuclides. These properties for radionuclides include half-life, radiotoxicity (or specific activity), and mobility in underground medium. Based on the review results, we proposed radionuclides selection process to prioritize the importance of radionuclides in the KAERI safety assessment.

Keywords: Radionuclides selection, Screening process, Safety assessment, Geological disposal, Spent nuclear fuel

Bayesian Updating in the Nuclide Travel Length for the Deep Borehole Spent Fuel Repository

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A GoldSim Total System Performance Assessment has been developed and utilized for assessment of the various conceptual HLW repositories for spent nuclear fuels during last a few decades. Even though, almost all required parameter values associated with the repository system are frequently assumed or sometimes overestimated, they are still far from being highly reliable. Uncertainties nested in nuclide transport modeling around the repository are mainly dominated by these parametric uncertainties aside from intrinsic model uncertainty. Reliable estimate of the parameter values commonly expressed as probability density functions (PDFs) always require a large amount of measured data. Such input distributions are used as input to the probabilistic assessment program through Monte Carlo simulation to quantitatively provide possible uncertainty of the results. However, in most cases, especially in the safety assessment of the repository which is typically related with both long-time span and wide modeling domain, inefficient observed data from the field measurements are common, making conventional probabilistic calculations rather even uncertain. Since Bayesian approach is known to be especially powerful and efficient in the case of lacking of available data measured, such short data could be compensated by coupling with a priori belief, reducing uncertainty. By allowing the a priori knowledge for incorporating insufficient observed data, which include expert' elicitation, their beliefs and judgment regarding the parameters as well as recent site-specific measurements, based on the Bayes' theorem, the older parameter distributions, "prior" distribution can be updated to a rather newer and reliable "posterior" distribution. Newer distributions are not necessarily expressed as PDFs for probabilistic calculation. These updates could be done even iteratively as many times as data values are sequentially available, which calls sequential Bayesian updating, making belief of posterior distributions become much higher by reducing parametric uncertainty. To show a possible way to enhance the belief as well as to reduce the uncertainty involved in parameter for the Bayesian scheme, nuclide travel length in the far-field area of a hypothetical deep borehole spent fuel Repository was investigated. The algorithm and module that have been developed and implemented in GSTSPA through current study was shown to work well for all assumed prior, three sequential posterior distributions and likelihoods.

Keywords: GoldSim, Deep borehole disposal, Safety assessment, Spent nuclear fuel, Bayesian updating, Travel length

Overseas Regulations and Cases About Natural Barrier for Deep Geological Disposal of High-Level Radioactive Waste

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The natural barrier, a component of the deep disposal system, has site-specific characteristics depending on the site of the repository, and is one of the main considerations for long-term safety evaluation after closure along with the engineered barrier among the multiple barrier systems of the repository. The natural barrier is defined in Korea as the natural underground and surface structures that can restrict the exposure of radioactive waste, human intrusion or groundwater infiltration into a disposal facility, and the transfer of radionuclides. It includes bedrocks and soils surrounding the engineered barriers of radioactive wastes [Notice of the NSSC, No. 2020021]. This study analyzed foreign regulatory requirements related to natural barriers, requirements for natural barrier and performance target of Sweden and Finland (safety functions and target characteristics of natural barriers, e.g. natural barrier composition, geological characteristics, hydrogeological characteristics). Overseas regulations and cases referenced to derive regulations of general safety requirements on natural barrier are IAEA SSG-14, SSMFS 2008:21 in Sweden, STUK/Y/4/2018 in Finland, and POSIVA SKB Report 01, a joint report between POSIVA and SKB. The repository site and repository depth should be chosen so that the geological formation provides adequately stable and favorable conditions to ensure that the repository barriers perform as intended over a sufficient period of time. The conditions intended primarily concern temperature- related, hydrological, mechanical (for example, rock mechanics and seismology) and chemical (geochemistry, including groundwater chemistry) factors. Furthermore, the repository site should be located at a secure distance from natural resources exploited today or which may be exploited in the future [SSMFS 2008:21]. Finland regulations also suggests similar requirements [STUK Y-4-2018]. According to the above regulations, POSIVA SKB report 01 mentions both the host rock and the underground opening as natural barriers and requires a safety function, and the main safety functions of the host rock and underground opening are as follows: (1) Isolation from the surface environment; (2) Favorable thermal conditions; (3) Mechanically stable conditions; (4) Chemically favorable conditions; and (5) Favorable hydrogeological conditions with limited transport of solutes. Such safety functions would provide insight for understanding of the natural barrier of deep geological disposal system.

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Keywords: High-level radioactive waste, Deep geological disposal, Natural barrier, Regulation

Numerical Comparisons for Assessing the Thermo-hydro-mechanical Simulation Performance of TOUGH-3DEC

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Discontinuum-based numerical methods can contain the multiple discontinuities in a model and reflect the thermal, hydraulic and mechanical characteristics of discontinuities. Therefore, discontinuum methods can be appropriate to simulate the model which require the detailed analysis of the coupled thermo-hydro-mechanical processes in fractured rock such as geothermal energy, CO₂ geo-sequestration, and geological repository of the high-level radioactive waste. TOUGH-3DEC, the three-dimensional discontinuum simulators for the coupled thermo-hydro-mechanical analysis, was developed by linking the integral finite difference method TOUGH2 and the explicit distinct element method 3DEC to describe the coupled thermo-hydro-mechanical processes in both porous media and discontinuity. TOUGH2 handles thermo-hydraulic analysis by the internal simulation module, and 3DEC performs mechanical study based on the constitutive models of porous media and discontinuity with coupling the thermal and hydraulic response from TOUGH2. The thermal and hydraulic couplings are the key processes and should be carefully verified by sufficient cases, so this study performed the thermo-mechanical and hydro-mechanical simulations which are modelling the analytic solutions including the uniaxial consolidation, fracture static opening, and the heating of a hollow cylinder problems. Each thermo-mechanical and hydro-mechanical verification case is also validated by comparing with the results of the other continuum and discontinuum-based numerical methods. TOUGH-3DEC results follow the analytic solutions and show better accuracy than the continuum-based numerical methods in the static fracture opening problem. The developed TOUGH-3DEC simulator can be expanded to coupled thermo-hydro-mechanical-chemical analysis in fractured rock mass, and the simulator needs to be verified by more complicated coupled processes problems which require in the chemical coupling.

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Keywords: Numerical simulation, Coupled processes, Discontinuum, Fractured Rock, Geological Repository

Study on the Release of Bentonite Colloids by Clay Chip-based Microfluidics

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Colloid-facilitated migration has been significantly concerned with the acceleration of the radionuclide mobility in the HLW repository. In the repository system, the compacted bentonite, which is the buffer material, could be the major source for colloid generation; hence, the understanding of colloid generation from the bentonite is the essential to expect the colloid-facilitated radionuclide migration. This study aimed to investigate the colloid generation using a bentonite-based micro-scale flow path system, which called microfluidics. In order to fabricate the microfluidics, direct milling method was applied to make a mold by computer numerical control. The fabricated mold applied to prepare the microfluidic chip by Polydimethylsiloxane (PDMS), in which the size of microchannel was designed to be one micrometer. Initially, sylgard 184 and curing agent mixed and stirred for 10 min, afterwards the bubbles in the paste was removed in the vacuum desiccator for 30 min. Then the paste was poured into the mold, and finally dried for 4 hours at 80°C in a dry oven. The compacted Ca-bentonite chip was prepared by the cold isostatic pressing (CIP) method with the dry density of $1.6 \text{ g}\cdot\text{cm}^{-3}$. The microfluidic chip and compacted bentonite chip were assembled by an acryl jig, the flow rate was adjusted by 20 mL syringe equipped syringe pump. The degree of colloid generation accompanied with the erosion of bentonite was gravimetrically examined after the experiment. The effect of the pH and ionic strength on the colloid formation was investigated through the particle size, stability and aggregation. To the best of our knowledge, this is the first examination for the colloid generation using microfluidics; these results would give information to understand the colloid formation from the compacted Ca-bentonite in the HLW repository system.

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Keywords: Microfluidics, Bentonite, Pseudo colloid, Colloid-facilitated migration, Erosion

Literature Review of Acceleration Tests for Nuclear Waste Repository Monitoring Sensors

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High level nuclear waste (HLW) is surely disposed in repository in safe by being separated from human life zone. Deep geological disposal method is one of the most potent disposal method. Deep geological repository is exposed to high pressure and groundwater saturation due to its depth over 500 m. And it is also exposed to high temperature and radiation by spent fuels. Thus, HLW repository suffers extremely complex thermo-hydro-mechanical-radioactive condition. Long-term integrity of repository should be verified because the expected lifetime of the repository is over 10,000 years. However, the integrity of monitoring sensors are not reach the endurance lifetime of the repository with present technology. And the disposal condition, thermo-hydro-mechanical-radioactive, should shorten the estimated lifetime of the monitoring sensors. Therefore, it is necessary to improve the long-term integrity of the monitoring sensors. Although long-term tests are required to identify the prolonged durability of monitoring sensors, accelerated tests can help curtail test period.

Accelerated tests is classified into accelerated stress test and accelerated degradation test and their methodology and theories are investigated. Their tests are design and proceed by following process: 1) identify failure modes, 2) select accelerated stress parameter, 3) Determine stress level, 4) Determine testing time and number of specimens, 5) Define measurement parameter and failure criteria, 6) Suggest measurement method and measurement duration. Literature reviews were conducted to identify the influence of the disposal conditions such as thermo-hydro-mechanical-radioactive on integrity of material and monitoring sensors. The investigated data reported in this paper will be utilized to verify the improvement of integrity of monitoring sensors.

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Keywords: Radioactive waste disposal repository, Thermo-hydro-mechanical-radioactive disposal condition, Long-term integrity of monitoring sensor, Accelerated test

Application of Coupled Hydro-mechanical Model to Numerical Simulation of 1-D Gas Injection Test

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Gases such as hydrogen can generate from the disposal canister in high-level radioactive waste disposal systems owing to the corrosion of copper container in anoxic conditions. The gas can be accumulated in the voids of bentonite buffer around the disposal canister if gas generation rates become larger than the gas diffusion rate of bentonite buffer with the low-permeability. Continuous gas accumulations result in the increase in gas pressure, causing sudden dilation flow of gases with the gas pressure exceeding the gas breakthrough pressure. Given that the gas dilation flow can cause radionuclide leakage out of the engineered barrier system, it is necessary to consider possible damages affected by the radionuclide leakage and to properly understand the complicated behaviors of gas flow in the bentonite buffer with low permeability. In this study, the coupled hydro-mechanical model combined with the damage model that considers two-phase fluid flow and changes in hydraulic properties affected by mechanical deformations is applied to numerical simulations of 1-D gas injection test on saturated bentonite samples (refer to DECOVALEX-2019 Task A Stage 1A). To simulate the mechanical behavior of microcracks which occur due to the dilation flow caused by increase in gas pressure, a concept of elastic damage constitutive law is considered in the coupled hydro-mechanical model. When the TOUGH-FLAC coupling-based model proposed in this study is applied, changes in hydraulic properties affected by mechanical deformations combined with the mechanical damage are appropriately considered, and changes in gas injection pressure, pore pressures at radial filters and outlet, and stress recorded during the gas injection test are accurately simulated.

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Keywords: Gas injection, Bentonite, Gas migration, Coupled hydro-mechanical model, Mechanical damage model

Analysis on the Long-term Evolution Phenomena Expected in the Korean Peninsula

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Since it takes hundreds of thousands of years for the radiotoxicity of spent nuclear fuel to decrease to natural levels, interactions between each repository barrier, climate change, and geological evolutions are inevitable. These processes should be defined as the long-term evolution FEPs and considered in the performance assessment to ensure the long-term safety of the disposal system. The literature survey on geological characteristics and history of the Korean peninsula was conducted, and the list of A-KRS-FEPs which are directly or indirectly related to long-term evolutions was identified in this study. The ice age and geological change are the capital phenomena considered in the exceedingly long-term evolution before/after climate change. The historical data on ice sheets and permafrost were analyzed to investigate the effects of the ice ages on the Korean peninsula. The sea-level changes were investigated based on the research on the coastal terrace to identify the impact on uplift and shoreline change accompanying the ice age. Also, the survey on the geological history data was conducted from the perspective of tectonic activity, metamorphism, igneous activity, and seismic activities to consider the geodynamic evolution of the Korean peninsula. As results, it was suggested that 14 FEPs were directly related to climate change, 18 FEPs were directly related to geological evolution, and 47 FEPs were indirectly relevant to long-term geodynamics. The consent-based FEPs and scenarios for the long-term evolution will be developed shortly, including most of the critical long-term evolution phenomena defined in this study and which are highly probable in domestic disposal conditions. The evaluation and verification of the APro system for long-term safety will accomplish using these FEPs and scenarios.

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

Hydrogeochemical Evolution Process of Granite Aquifer in and Around Underground Research Tunnel

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The geological disposal of spent nuclear fuel is one of the important problems to be solved worldwide. For the safety of the geological disposal, disposal facility is recommended to be constructed in the deep reducing environment of host rocks. As host rocks, rock salt, argillaceous (clay) rock, and crystalline rock have been considered as stable geological formations in various countries. Although various studies have been conducted on crystalline rocks in Korea, there are still few studies on hydrogeochemical evolution in the deep and reducing environment related to the disposal of spent nuclear fuel. Therefore, this study was conducted to identify hydrogeochemical evolution process in granite aquifer which can affect the stability of disposal facility. Groundwater samples for isotope and chemical analysis were collected quarterly adjacent to KURT (KAERI Underground Research Tunnel). As the depth increased, the groundwater changed from Ca-HCO_3 type to Na-HCO_3 type under the influence of silicate mineral weathering, and the fluorine concentration increased due to the dissolution of fluorine-bearing minerals. However, hydrogeochemical evolution according to the depth was not observed in some wells because of a hydraulic connection through the fracture zone. In addition, the behavior of nitrate and redox-sensitive metals (Fe, Mn, U, Mo) in groundwater was clearly different in the redox condition. Considering these hydrogeochemical processes and hydrogeological factors, a conceptual model of granite aquifers in and around KURT was established. The results of this study will be used as basic data to understand the hydrogeochemical processes and to evaluate and predict the behavior of radionuclides in granite aquifer system.

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Keywords: Radwaste disposal, Water-rock interaction, Hydrogeochemical evolution, Silicate weathering, Redox condition

Artificial Intelligence Based Effective Signal Discrimination Technology for Structure Monitoring and Diagnosis

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In order to monitor the long-term condition of structures in nuclear waste disposal system and evaluate the degree of damage, it is necessary to secure quantitative monitoring, diagnosis, and prediction technology. However, at present, only simple monitoring or deterioration evaluation of the structure is being performed. Recently, there is a trend to develop monitoring systems using artificial intelligence algorithms, such as to introduce artificial intelligence-based failure diagnosis technology in nuclear power plant facilities. An artificial intelligence algorithm was applied to distinguish the noise signal and the destructive signal collected in the field. This can minimize false alarms in the monitoring system. However, it is difficult to apply artificial intelligence to industrial sites only by learning through laboratory data. Therefore, a database of noise signals and destructive signals was constructed through laboratory data, and signals effective for quantitative soundness determination of structures were separated and learned. In addition, an adaptive artificial intelligence algorithm was developed to enable additional learning and adaptive learning using field data, and its performance was verified through experiments.

Acknowledgements

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Keywords: Structures in nuclear waste disposal system, Structure monitoring and diagnosis, Adaptive artificial intelligence, Effective signal discrimination

Dual-structure of Expansive Clays and its Response to Hydro-mechanical Loadings: A Review

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Expansive clays (for examples, bentonites) are favored as buffer and backfill materials because of their low hydraulic conductivity, high swelling potential, and good mechanical properties, and are installed in highly compacted blocks in repositories. Compacted expansive clays have a dual-structure system: macrostructural system which is a complex of clay aggregates with the inter-aggregate pores (macropores) which can be filled by either liquids or gases; microstructural system with the intra-aggregate pores between or within clay particles (micropores) which is usually considered to be saturated by liquid. Understanding the dual-structure system of expansive clays is essential for characterizing and modeling multiphysics (stress-strain, swelling pressure, etc.) in buffers and backfills.

Existing multiphysics studies of expansive clays, as in non-expansive soils, were mostly conducted with a single structure approach based on the behavior of macropores, and there have been limitations in the comprehensive interpretation and modeling of experimental results. However, with the recent development of measurement techniques, a lot of available information on the pore structure of compacted expansive clays has been reported, and with the results, a dual-structure approach considering both microstructural and macrostructural systems has been increasingly applied to improve the modeling of multiphysics of expansive clays.

This study reviewed the dual-structure system of compacted expansive clays, analyzed previous studies on its evolution according to hydromechanical loading (loading-unloading and wetting-drying paths), and based on these, intended to provide technical knowledge and information needed for multiphysics research of expansive clays-based buffer and backfill for the KRS repository.

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Keywords: Dual-structure, Hydromechanical behavior, Expansive clays, Buffer, Backfill, High-level repository

A Case Study of Application of Natural Analogues to Geochemical Tools Focusing on Complex Transport Behavior of Radionuclides

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Several previous simulation studies using various geochemical models have been carried out in several major analogue sites. The cases are beneficial when these studies provided the possibility of testing the geochemical models to be used to describe the migration of radionuclides in a future radioactive waste repository system. It was possible to interpret the complex transport behaviour of radionuclides such as uranium and thorium in an environment. We organize major natural analogue study sites from the previous literatures that provided information on the general geochemistry of the sites, in terms of groundwater composition and mineralogy. Also, we calculated aqueous speciation and the solid phases most likely to control their solubilities. The results obtained from the previous studies and this study vary depending on the tools used and on the conceptual models followed. Also, the results differed from the actual measured concentrations of trace metals or radionuclide analogues. The results obtained from these tests identify the main mathematical limitations of available geochemical models. However, the modelling results using a geochemical code with the thermodynamic database simulated well the observed behaviour of radionuclides, especially to identify the dominant processes controlling actinide mobilization and fixation. It was a useful outcome in terms of building confidence on the current geochemical tools to predict the concentrations of radionuclide analogues once the major geochemical characteristics were known. This study allows improving specific aspects of geochemical modelling using major natural analogue sites.

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Keywords: Natural analogue, Geochemical modelling, Radionuclides, Migration

Development of Biosphere Assessment Module Considering the Long-Term Evolution of the Surface Environment

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The safety assessment of a geological disposal system is performed over a period of hundreds of thousands of years, during which the activity of radionuclides in spent nuclear fuel decreases to natural radioactivity levels. During this period, the biosphere also experiences the long-term evolution of the surface environment including climate, terrain, and ecosystem changes. These changes cause changes in the water balance, which in turn change the pathways of radionuclides in the subsurface. Therefore, it is essential to consider these long-term changes in the surface environment for a reasonable biosphere safety assessment. For this purpose, this study developed the biosphere assessment module considering the long-term evolution of the surface environment, as a sub-module of APro (Adaptive process-based total system performance assessment framework). As a preceding study, the biosphere assessment module was previously developed using COMSOL for hydraulic and radionuclide transport processes, to simulate the pathway of radionuclides traveling from the shallow aquifer to the surface water body and soil. To consider the long-term evolution of the surface environment, the previous module needed to be improved to apply different water balances as boundary conditions of the module at each snapshot, which is a sub-time period divided based on the surface evolution data. To this end, this study utilized SWAT (Soil and Water Assessment Tool) which calculates the water balance using the surface environmental data including climate, terrain, land cover, and soil type. Conceptually, SWAT calculated annual water balance considering surface environmental changes, and certain components (i.e., groundwater recharge and hydraulic head of water bodies) of water balance were transferred to COMSOL as external data to simulate the pathway of radionuclide transport and spatio-temporal variability of radionuclides. At the current stage, the biosphere computational module has been developed to correspond to its conceptual model, and we plan to further test the applicability of the module using different surface environmental data.

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Keywords: Biosphere assessment, Radionuclide transport, Geological disposal system, COMSOL, SWAT

Measurements of Thermal Conductivity of Compacted Sand-bentonite Mixtures

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The buffer material plays a role in preventing the excessive rise in temperature generated from the high-level radioactive waste by dissipating the decay heat to the rock. For this reason, the buffer material must have thermal properties to ensure the performance of the deep geological repository. This study measured the thermal conductivity of sand-bentonite according to the mixing ratio to improve the thermal properties. The compacted buffer was manufactured with a sand-bentonite mixing ratio of 6:4, 7:3, and 8:2 with 9 to 12% water content. As a result, the thermal conductivity increases as the ratio of sand increases. As a further study, it is necessary to experiment on whether sand-bentonite's hydraulic, mechanical, and chemical performance is suitable for the stable operation of a repository.

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Keywords: Sand-bentonite, Thermal conductivity, Buffer material, Deep disposal repository

Quantitative Analysis of IFEP for the Development of Korean High-level Waste Repository

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Safety evaluation of high-level radioactive waste disposal facilities including spent nuclear fuel is a very urgent and critical issue, and in order to do so, it is very important to develop a safety case that includes Feature, Event, Process (FEP) analysis, scenario development, and scenario uncertainty evaluation. In the case of Korea, the disposal of spent nuclear fuel is recognized as an unavoidable option, and in the end, Korea's specific FEP (SFEP) development and safety evaluation according to the scenario should be conducted. Because each country's situation and environment are different, it is necessary to develop an SFEP based on a generic FEP (International FEP). To this end, an understanding of IFEP is essential. In this study, about 1,000 major terms appearing in the OECD/NEA IFEP are classified to where each of them belongs among F, E, and P, and which FEP each word belongs to, and the correlation between the frequency of occurrence and each term is analyzed. This result will serve as a reference for the results of SFEP analysis such as POSIVA and SKB, which our research team will analyze later. In addition, each term belongs to which academic field, and the most appropriate translation for translating each term into Korean is also described.

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Keywords: Spent nuclear fuel, Disposal, FEP, Safety case, FEP analysis

4분과

중저준위폐기물관리(Oral)

Development of Compton CT for Radioactive Waste Drum Monitoring: Optimization and Performance Prediction

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During decommissioning of a nuclear power plant, a large amount of radioactive waste is produced, and it is known to cost more than 300 billion won to dispose the waste. To reduce the disposal cost, it is essential to minimize the number of radioactive waste drums, which can be achieved by detecting and removing hotspot contaminations in the radioactive waste drums. Therefore, a Compton CT system for radioactive waste monitoring is under development, which provides the images of both the internal structure of the drum and the radioactive hotspot(s) in the drum. Based on the acquired information, the activity of hotspots can be estimated. The performance of the system is affected by various geometry factors. Therefore, it is essential to determine optimal configuration by evaluating the effects of the factors on the performance of the system. In the present study, we determined the optimum value of the factors and then predicted the performance of the optimized system by using a simulator based on the Geant4 Monte Carlo simulation. For optimization, the factors were evaluated in terms of structural similarity index measure (SSIM) and measurement time. The considered factors were the activity of the CT source, source to object distance (SOD), object to detector distance (ODD), and projection angle. The simulation result showed that the activities of the CT sources were determined as 23 mCi for ^{137}Cs and 9.6 mCi for ^{60}Co . The optimal SOD and ODD were 180 cm and 40 cm, respectively. The optimal projection angle was evaluated as 4° since it achieves the SSIM of 0.95 faster than other projection angles. With the optimized parameters, the performance of the system was evaluated using the IAEA gamma CT standard phantom containing a hotspot of ^{137}Cs (7.02 μCi). The Compton image was reconstructed using the back-projection algorithm, and the CT image was reconstructed using the filtered back-projection algorithm. The result showed that the location of the hotspot in the Compton image was well identified at the true position. The acquired CT image also well represented the internal structure of the phantom, and the estimated mean linear attenuation coefficient value ($\mu = 0.0789 \text{ cm}^{-1}$) of the phantom was close to the true value ($\mu = 0.0752 \text{ cm}^{-1}$). In addition, the hotspot activity estimated by combining the information of the Compton image and CT image was 8.06 μCi . Hence, it was found that the Compton CT system provides essential information for radioactive waste drums.

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Keywords: Compton CT system, Radioactive waste drum, Gamma CT, Compton imaging, Activity estimation

Development of Condition Monitoring Technology for Safe Management of Radioactive Waste Drum Monitoring

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With the development of the nuclear industry and the increase in the use of radioactive materials, the generation of radioactive waste is increasing. As the generation of radioactive waste increases, the occurrence of related safety accidents is also increasing, and it is necessary to develop a radioactive waste monitoring technology to prevent such accidents in advance and efficiently manage radioactive waste.

In Information and Communication Technology (ICT), various ICT technologies such as Internet of Things (IoT), Augmented Reality (AR), and Virtual Reality (VR) that can help with the safety management of these radioactive wastes are being developed. In this study, a radioactive waste monitoring technology was developed using ICT technology, such as management of the entire cycle history of waste using Quick Response (QR) codes, and development of AR visualization technology for small packages of radioactive waste. In addition, by using IoT technology to collect desired data from sensors and store the results, after the waste drum is loaded in the waste storage, a technology was developed to track and monitor the history and movement of the waste drum from repackaging to transfer to the storage.

The data required for monitoring the radioactive waste drum includes location information, whether the drum is open or closed, temperature and humidity, etc. To collect this information, a drum monitoring technology was built with a 2.4 G wireless router, an anchor constituting a virtual zone, a tag to be mounted on the drum container, and a WNT server that collects sensor data. The network tool provided by WirePas was used for network configuration, and the status of gateways and nodes can be monitored by interworking with the WNT server. The configured IoT sensor technology were tested in a waste storage environment. Four anchors were installed and linked to the network to match the virtual zone and the real storage zone, and it was confirmed whether the movement of the tag was recorded on the network while moving the tag including the IoT sensor for analyzing location information. Based on these research results, it can contribute to the safety management of radioactive waste and establishment of Waste Acceptance Criteria (WCP) by and managing the history and monitoring the waste in the entire cycle from repackaging to disposal.

Keywords: Radioactive waste, Status monitoring, Safety management, IoT sensor

Application of Bayesian Statistics to Infer Radioactivity of Uranium Wastes

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A method of quantitatively analyzing radioactivity of uranium waste in the In-situ measurement using Bayesian inference was proposed. When applying the traditional efficiency calibration method, which uses standard sources or Monte Carlo simulation, the radioactivity error is large depending on the degree of spread of the radioactive contamination especially in large sample such as a 200 L drum. In addition, the existing method has a limitation in that it is difficult to reflect the uncertainty according to the location of the source. In this preliminary study, to overcome the limitations of the existing method, a Bayesian statistical-based radioactivity quantitative analysis model was proposed that can increase the accuracy of analysis even in situations where radioactive contamination of uranium waste is non-uniformly distributed. As a result of evaluating the simulated waste with the proposed Bayesian method, the accuracy was improved more than about 6 times compared to the classical efficiency calibration method.

Keywords: Bayesian inference, Uranium activity, In-situ measurement, Radiowastes, Non-uniform

Desing and Development of Monitoring System for Near-surface Disposal Facility Engineering Barrier

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The disposing method of the low-intermediate-level radioactive waste, near-surface disposal facilities are generally used. This disposal method refers to a method of constructing a concrete structure on the surface of the ground, putting radioactive waste in it, and covering it with an engineered barrier to isolate human life. Among these, engineered barriers mean covering multiple layers of heterogeneous materials such as sand, clay, and gravel. Engineering barriers have the purpose of delaying the release of radioactive materials into the natural environment as much as possible, and maintaining the isolation of radioactive waste and human life for as long as possible. In this study, the design and construction method of the facility to demonstrate the performance of the engineered barrier that isolates the surface disposal facility from nature was described. In addition, the design and construction method of monitoring technology that can monitor the safety of engineered barriers by measuring information such as moisture, temperature, and slope safety in real time was also explained.

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Keywords: Radioactive waste, Near-surface disposal, Engineering barrier, Monitoring system

Evaluated Cs Pollucite Ceramic Waste Form According to Mixing Ratio of Matrix Materials

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Radioactive Cesium is fission products of spent nuclear fuel with high heat generating nuclide, having a 30 years half-life. Particularly, it is important to make stable waste form because Cs-137 have high solubility and mobility at ground water. The ceramic waste form has higher thermal and structural stability and lower solubility than glass and cement waste form. Various ceramic waste forms for Cs immobilization have been researched such as aluminosilicate ($\text{CsAlSi}_2\text{O}_6$), phosphate ($\text{CsZr}_2(\text{PO}_4)_3$), titanate ($\text{Cs}_x\text{Al}_x\text{Ti}_{8-x}\text{O}_{16}$) and $\text{CsZr}_{0.4}\text{W}_{1.5}\text{O}_6$. Cs pollucite is incorporated radio-Cesium to aluminosilicate framework by inorganic ion-exchange with zeolite. Therefore, it is an extremely stable structure. In previous study, we are prepared Cs pollucite pellet with various ratio of Cs precursor/matrix materials, and attempted to evaluate applicability as ceramic waste form. Cs pollucite is produced by mixing Mullite and SiO_2 obtained by heat treatment Kaolinite with Cs_2CO_3 in ratios of 0.5, 0.6, 0.7, 0.8. Optimized ratio was 0.5 revealed single pollucite phase and the others exhibited CsAlSiO_4 phase with pollucite. Cs pollucite of ratio 0.5 was pelletized under various conditions and evaluated performance as waste form. herein, the pellets were cracked on surface and edges broken. Therefore, Cs pollucite having high ratio of matrix materials contained Si and Al was prepared and pelletized, and then waste form was evaluated. The Cs pollucite powder is ratio of Cs precursor/matrix materials were 0.1, 0.2, 0.3, 0.4. Pollucite powder was mixed with 1.5, 2.0wt% Polyvinyl alcohol as binder, and dried at 70°C for overnight. Afterward, these powders obtained were pressed using punch-die apparatus at 50, 100 bar for 1 hour and the pellets with about dia. 25 mm and height 10 mm was acquired. These pellets were sintered at 1,400°C for 5 hours. Subsequently, the waste forms were evaluated physicochemical test such as compression strength, thermal conductivity, thermal expansion and leaching properties analysis.

Keywords: Radioactive cesium, Cesium pollucite, Ceramic waste form, Pelletization

Influence of Si/Al Ratio and Curing Temperature of Geopolymer Waste Forms on Immobilization of Radioactive Borate Waste

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Immobilization of radioactive borate waste containing a high boron concentration using cement waste form has been challenged because the soluble borate phase such as boric acid reacts with calcium compounds, hindering the hydration reaction in cement waste form. Metakaolin-based geopolymer waste form which has a pure aluminosilicate system without calcium can be a promising alternative for the cement; however, secondary B-O-Si networks are formed by a reaction between borate and silicate, resulting in poor mechanical characteristics such as low compressive strength and final setting retardation. Thus, it is important to optimize the Si/Al molar ratio and curing temperature which are critical parameters of geopolymer waste form to increase borate waste loading and enhance the durability of geopolymer. Here, metakaolin-based geopolymer waste form to immobilize simulant radioactive borate waste was fabricated by varying the Si/Al molar ratio and curing temperature. The 7 days-compressive strength results reveals that the Si/Al molar ratio of 1.4 and curing at 60°C is advantageous to achieving high waste loading (30wt%). In addition, geopolymer waste forms with the highest borate waste loading exceeded the 3.445 MPa after the waste form acceptance criteria such as thermal cycling, gamma irradiation, and water immersion tests. The leachability index of boron was 7.56 and the controlling leaching mechanism was diffusion. The thermal cycling and gamma irradiation did not significantly change the geopolymer structure. The physically incorporated borate waste was leached out from geopolymer waste form during leaching and water immersion tests. Considering these results, metakaolin-based geopolymer waste form with a low Si/Al ratio is a promising candidate for borate waste immobilization, which has been difficult using cement.

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Keywords: Radioactive waste, Borate waste, Geopolymer, Si/Al ratio, Waste form acceptance criteria

Development of High Intensity Focused Ultrasonic to Decontaminate Complex Metal Wastes

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High-intensity focused ultrasonic (HIFU) decontamination technology to decontaminate complex metal radioactive waste was developed and verified. Ultrasonic decontamination technology is a method widely used in this field, but its energy strength is weak, so it cannot be applied to fixed contamination. The HIFU developed in this study can eliminate a wide range of fixed contamination due to the advantage of maintaining a high frequency while having hundreds of times the energy intensity compared to conventional general ultrasonic method. In addition, there is a merit in that there is no work that generates a lot of secondary wastes such as chemical decontamination method or threatens the safety of workers. In particular, high ultrasonic energy is transmitted to curved parts and inside pipes that cannot be decontaminated with blasting method, so various types of metal wastes can be treated with the HIFU method.

In this study, the performance of the HIFU was verified for zirconium chips, and the radioactivity after decontamination was reduced to less than MDA in all subjects.

Keywords: High intensity ultrasonic, Decontamination, Radiowastes, Uranium, Metal wastes

Degradation of Re(IV)-EDTA in the Environment as Analogue of ^{99}Tc -EDTA for Safe Disposal in the Geological Repository

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Mechanism and kinetics of Rhenium complexes as a surrogate of Technetium-99 (^{99}Tc) is worthy of study from radioactive waste safe disposal perspective. Re(IV)-EDTA was synthesized via the reduction of Re(VII) with Sn(II) in the presence of Ethylenediaminetetracetic acid (EDTA). The Re(IV)-EDTA was then degraded by H_2O_2 (7–30%) at pH of 3–11 in ionic strength $I = 0\text{--}2\text{ M}$ solution. The Re-EDTA was observed to degrade more rapidly at pH of $\leq 3\text{--}4$ than one of $\geq 10\text{--}11$ and remained stable at pH = 7–9. At a low acidic pH, the complex degradation process was facilitated by protonation and corresponded to the exponential model ($y = k \cdot e^{-nt}$). In contrast, at an alkaline pH, the degradation was facilitated OH^- complexation with Re(IV) and corresponded to a linear model ($y = -mt + C$). Complex degradation followed the zero-order rate kinetics for the H^+ and Re-EDTA parameters, apart from a pH of 3, for which degradation was a better fit to first order kinetics. A higher Re(IV)-EDTA stability at a pH of 7–9 demonstrated that Re(IV)-EDTA (or $^{99}\text{Tc(IV)-EDTA}$) tends to be more persistent in natural environmental conditions.

Keywords: Reductive coprecipitation, Tc mobility, Environmental release, Radionuclide release, Oxidation-induced mobility, Tc(7)/Tc(4)

Consideration on Cellulose Disposal Limits in Near Surface Radioactive Waste Disposal Facilities

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It has been discovered that the isosaccharinic acid (ISA) formed in a cellulose degradation leachate were capable of forming soluble complexes with thorium, uranium (IV) and plutonium. Since 1993, the ISA has received particular attention in the literature due to its ability to complex a range of radionuclides, potentially affecting the migration of radionuclides.

ISA is formed as a result of interactions between cellulosic materials within the waste inventory and the alkalinity resulting from the use of cementitious materials in the construction of the repository.

In an alkaline cementitious environment, cellulose degrades mainly via a peeling-off reaction. The main degradation product is ISA, a polyhydroxy type of ligand forming stable complexes with tri- and tetravalent radionuclides. ISA can have an adverse effect on the sorption of radionuclides to an extent which depends on its concentration in the cement pore water and potentially enhance their mobility. The concentration of ISA is governed by several factors such as cellulose loading, cement porosity, extent of cellulose degradation, etc. The sorption of ISA on cement, however, is the process which governs the concentration of ISA in the pore water. According to the experimental result from a literature, the ISA concentration in facilities with a cellulose loading of 5% is calculated to be of the order of 10^{-4} M. At this level, the effect of cellulose degradation products on radionuclide sorption is negligibly small.

Recently in Korea, cellulose limits as waste acceptance criteria is studying and planning to prepare the detailed requirement for near surface radioactive waste disposal facilities. It is desirable to suggest consideration on cellulose disposal limits around the time that the regulatory body and concern organizations establish the cellulose disposal limits as follows. Firstly, identify the cellulose effect on the sorption of the nuclides as cementitious disposal environments such as affected nuclides, threshold value and contribution to radiological risks under domestic disposal environment. Secondly, make sure and consider the difference between lab-scale experimental conditions and probability occurring in real disposal conditions such as probability for generation and persistence of pH in cellulosic material disposal conditions and cellulosic material disposal methods. Finally, consider characterization of cellulosic material such as polymerization, contents of cellulose in law material and time of degradation process.

As a result, desirable cellulose limits are to set up for both safety and economic aspect.

Keywords: Cellulose, Disposal, Waste acceptance criteria, LILW

Preliminary Safety Evaluation for Cellulose Disposal at 1st Phase Disposal Facility

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Recently, concern regarding disposal of cellulosic material is growing as cellulose is known to produce complexing agent, isosaccharinic acid (ISA), upon degradation. ISA could enhance mobility of some radionuclides, thus increasing the amount of radionuclide released into the environment. Evaluation on the possible impact of the cellulose degradation would be an important aspect in safety evaluation. In this paper, the maximum safe disposal amount cellulose is evaluated considering the disposal environment of silos of 1st phase disposal facility.

The key factor governing the impact of cellulose degradation is pH of disposal environment, as cellulose is known to degrade partially at pH above 12.5, and completely at pH above 13. Thus, disposal environment should be analyzed as to determine the extent of degradation.

As silos are constructed with large amount of cement, porewater within concrete walls would be of very high pH. However, for high pH porewater to be released into the pores of crushed rock, which is filling up the silos, lower pH groundwater (commonly pH 7) should flow into the silos through the concrete walls. This causes dilution of the high pH concrete porewater, resulting in a lower pH as the silos are filled, reaching to expected pH of 11.8–12.3, which is below cellulose degradation condition. Thus, cellulose degradation is not expected, but to quantitatively evaluate safe disposal amount of cellulose, partial degradation is assumed.

Upon literature review, the most conservative ISA concentration, enhancing radionuclide mobility, is determined to be 1.0×10^{-4} M and to reach this concentration, cellulose mass equivalent to 6wt% of cement of the repository, is required to be degraded. However, this ratio is derived based on complete degradation of cellulose into ISA, so for partial degradation, degradation ratio and yield ratio of ISA should be considered.

Commonly, cellulosic material (e.g. cotton, paper, etc.) has degree of polymerization (DP) between 1,000–2,000, and with this DP, degradation ratio is estimated to be about 10%. Furthermore, yield ratio of ISA is known to be 80%. Considering all these aspects, about 1.79×10^7 kg of cellulose could be disposed, which if converted into number of drums, considering cellulose content of dry active waste, more than 100,000 drums (200 L) could be disposed with negligible impact on safety.

Based on the result, negligible impact of cellulose degradation is expected for safety of 1st phase disposal facility. In future, this study could be used as fundamental data for revising waste acceptance criteria.

Keywords: Cellulose degradation, 1st phase disposal facility, ISA, Safety evaluation

Safety Evaluation in Case of Accidents Handling Glass Fiber Radioactive Waste Using Q-system

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Glass fiber, which was used as an insulation material in pipes near the steam generator system of nuclear power plants, is brittle and the size of crushed particles is small, so glass fiber radioactive waste (GFRW) can cause exposure of workers through skin and breathing during transport and handling accidents.

In this study, Q-system which developed IAEA (International Atomic Energy Agency) for setting the limit of radioactivity in the package is used to confirm the risk of exposure due to an accident when transporting and handling GFRW. Also, the evaluated exposure dose was compared with the domestic legal effective dose limit to confirm safety.

Q-system is an evaluation method that can derive doses according to exposure pathway (EP) and radioactivity. Exposure doses are calculated by dividing into five EP: Q_A , Q_B , Q_C , Q_D , and Q_E . Since the Q-system is used to set the limit of radioactivity that the dose limits is satisfied to nearby workers even in package handling accidents, the following conservative assumptions were applied to each EP. Q_A , Q_B are external EP of assuming complete loss of package shielding by accident and radiation are received for 30 minutes at 1 m, Q_C is an internal EP that considers the fraction of nuclides released into the air and breathing rate during accident, and Q_D is an external EP that skin contamination for 5 hours. Finally, Q_E is an internal and external EP by inert gases (He, Ne, Ar, Kr, Xe, Rn) among the released gaseous nuclides, but the Q_E pathway was excluded from the evaluation because the corresponding nuclide was not present in the GFRW products used for evaluation.

In this study, the safety evaluation of GFRW was performed package shielding loss and radioactive material leakage due to single package accident according to assumption of four pathways, and the nuclide information used the average radioactivity for each nuclide of GFRW. As a result of the dose evaluation, Q_A was evaluated as 2.73×10^{-5} mSv, Q_B as 1.06×10^{-6} mSv, Q_C as 7.53×10^{-3} mSv, and Q_D as 2.10×10^{-6} mSv, respectively, and the total exposure dose was only 7.56×10^{-3} mSv, it was confirmed that when compared to the legal limits of the general public (1 mSv) and workers (20 mSv) 0.756% and 0.038%, respectively. In this study, it was confirmed that the legal limitations of the general public and workers were satisfied evens in the event of an accident as a result of evaluating the exposure dose of nearby targets for package shielding loss and radioactive material leakage while transporting GFRW. In the future, the types of accidents will be subdivided into falling, fire, and transportation, and detailed evaluation will be conducted by applying the resulting accident assumptions to the EP.

Keywords: Glass fiber, Q-system, Exposure pathways, Transporting and handling accident

Development of RUCAS: Verification of Dose Assessment Model for Recycling of Metallic Radioactive Waste

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In Korea, it is expected that the decommissioning of nuclear reactors will increase due to the license termination of reactors constructed in the 1960s to the 80s. According to the investigation of KORAD, VLLW accounts for 67.10% of decommissioning wastes and amounts to about 413,336 drums. Due to their huge amount, it is necessary to create an appropriate decommissioning waste management plan even though VLLW is disposed at the second-phase disposal facility of the Gyeongju repository. For efficient reduction in decommissioning wastes, it is required to actively use a clearance of metallic and concrete radioactive wastes. Regulations of nuclear safety and security commission notice that the radioactive waste can be reused or recycled if it meets the clearance criterion, $10 \mu\text{Sv}\cdot\text{y}^{-1}$ for individual dose. Therefore, it is important to develop a computational code which calculate individual doses for each scenario, and determine whether the clearance criterion is satisfied. However, in the case of metallic waste, RESRAD-RECYCLE used in dose assessment for the clearance has no longer been maintained or updated since 2005 and there is no code for recycling of concrete waste. For this reason, a dose assessment code RUCAS (Recycle-Underlying Computational dose Assessment System) has been developed by Ulsan National Institute of Science and Technology (UNIST). A point kernel method is adopted into external dose assessment model to calculate more realistic options, which are various geometries of source, and shielding effect. In the case of internal radiation, equations of internal dose from IAEA are used.

This research conducts a verification of dose assessment model for recycling of metallic radioactive waste. RESRAD-RECYCLE is the comparison object and results from RESRAD-RECYCLE validation report are referenced. Targets are 14 recycling scenarios composed up to the smelting metal step of four steps, which are arising scrap metal, smelting scrap metal, and fabrication of metal product, and reusing/recycling of product. Seven isotopes, which are Ac-227, Am-241, Co-60, Cs-137, Pu-239, Sr-90, and Zn-65, are selected for calculation. Validation results for external dose vary by isotopes, but show acceptable differences. It seems to be caused by difference in the calculation method. In the case of internal dose using same calculation formula, results are exactly matched to RESRAD-RECYCLE for all isotopes. Consequently, RUCAS can conduct functions supported by RESRAD-RECYCLE well and future work will be conducted related to domestic recycling scenarios considering public acceptance, and verification with radiation shielding codes for various geometries of source.

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Keywords: Exempt waste, Recycling, Computational code, Dose assessment

Review of Foreign Management and Disposal Policies for Intermediate-Level Waste

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When the decommissioning of a nuclear power plant begins in earnest, starting with Kori Unit 1, it is necessary to dispose of intermediate-level wastes such as high-dose waste filters and waste resin stored in the power plant, as well as the internal structures of the reactor. However, there are no intermediate-level waste disposal facilities in Korea, and the maintenance of acceptance criteria considering the physical, chemical, and radiological characteristics of intermediate-level waste is insufficient. In this paper, in preparation for the establishment of domestic intermediate-level waste treatment/disposal and acceptance standards, the following major foreign countries' legal and institutional standards for intermediate-level waste are reviewed, and based on this, factors to be considered when establishing domestic intermediate-level waste treatment/disposal standards were derived. First, although the USA does not define and manage intermediate-level wastes separately, low-level wastes were separated into Class A, B, and C, where land disposal is allowed, and GTCC, which does not allow land disposal. However, it was recently confirmed that the position was changed to recognize the possibility of land disposal of GTCC waste under the condition that the dose to inadvertent intruders does not exceed $5 \text{ mSv} \cdot \text{yr}^{-1}$ and a barrier against inadvertent intrusion valid for 500 years is installed. Second, Sweden classifies intermediate-level wastes into short-lived and long-lived intermediate-level wastes. The maximum dose rate permitted on packages are different for each vault and a silo of the SFR where short-lived wastes; $100 \text{ mSv} \cdot \text{h}^{-1}$ or less is disposed of in BMA, $10 \text{ mSv} \cdot \text{h}^{-1}$ or less in BTF, $2 \text{ mSv} \cdot \text{h}^{-1}$ or less in BLA and $500 \text{ mSv} \cdot \text{h}^{-1}$ or less in silo. Meanwhile, a repository for long-lived low and intermediate level waste, SFL, which could contains significant amounts of nuclides with a half-life greater than 31 years, operations are planned to commence in 2045. Third, France also manages short-lived intermediate-level wastes and long-lived intermediate-level wastes separately, and the short-lived intermediate-level wastes were disposed of together with short-lived low-level wastes at the La Manche and L'Aube repository. France announced the Cigéo Project, a high- and medium-level long-lived waste plan in 2012, and submitted the creation authorization application for in 2021 with the goal of operating a repository in 2025. Finally, the UK defines intermediate-level waste as "waste whose activity level exceeds the upper limit for low-level waste but does not require heating, which is considered in the design of storage or disposal facilities" and established NIREX to provide deep disposal of intermediate-level radioactive waste. In Finland, wastes with radioactive concentrations of 1 MBq/kg to $10 \text{ GBq} \cdot \text{kg}^{-1}$ are classified as intermediate-level wastes, and a repository was constructed and operated in a bedrock of about 110 m underground. Because the domestic classification standard simply classifies intermediate-level waste as waste exceeding the activity level of low-level waste limit, not high-level wastes, it is necessary to establish treatment and disposal standards by subdividing them by dose rate and long-lived radionuclides concentration to safely and efficiently dispose of intermediate-level waste for. Additionally, there is a need to decide whether or not to reflect safety by inadvertent intruders when evaluating the safety of intermediate-level disposal.

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Keywords: Intermediate-level, Waste, Disposal, GTCC, Policy

Long-term Chemical Stability Study for I Waste-form Using TAB (Te-Ag-Bi) Matrix in Domestic Underground Water

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During the treatment of spent nuclear fuel, radioactive iodine is generated in a liquefied or gaseous form in a specific process. In the case of iodine 129, it is a long-lived nuclide with a very long half-life and has high groundwater mobility under repository conditions. Despite showing a low radioactivity value, research on the management of radioactive iodine from a long-term perspective is continuously being performed. Although research has been conducted using borosilicate glass as a medium for solidifying iodine, compatibility of I in borosilicate glass is very small and the volatility is high in the solidification process. So it is not suitable as a solidified substance of iodine. Therefore, studies on other solidification media to replace them are continuously being conducted.

Our research team tried to develop a new medium that can contain iodine in a solidified body stably through a simple heat treatment process and can improve problems such as volatility and waste loading. Iodine is captured as AgI in the Ag ion-exchanged zeolite. So, TeO₂, Ag₂O, and Bi₂O₃ having a high AgI loading rate were used as main components. It was named TAB after taking the first letter of each element. In previous studies, the physical properties, structure, and chemical stability of TAB materials were confirmed. PCT (Product Consistent test) was performed to confirm chemical stability. It is mainly used to compare the chemical stability of glass materials with other glass materials, but there are limitations in evaluating the long-term chemical stability of materials. In this experiment, we tried to evaluate the long-term stability of TAB and compare it with borosilicate, which is conventionally used to treat radioactive waste. In addition, we tried to understand the leaching behavior inside the TAB medium. For this purpose, ASTM C1308 test was performed for 365 days, and distilled water and KURT groundwater were used as leachates to examine the effect of ions in the groundwater on the solidified body. To analyze the leaching behavior, ICP-MS and ICP-OES analyses were performed, and the cross-section of the sample after leaching was observed through SEM.

Keywords: I waste-form, Te-Ag-Bi, Long-term stability test, Ground water

4분과

중저준위폐기물관리(Poster)

Degradation of Tributyl Phosphate via Plasma-based Advanced Oxidation Process Using Submerged Multi-hole Dielectric Barrier Discharge

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Tributyl phosphate (TBP) is a well-known and important compound in the nuclear industry for the nuclear fuel reprocessing, and it is also used in a various field such as plastic industry as antifoaming agent. Untreated organic pollutants in TBP can remain in the soil water and cause serious environmental pollution, thus it should be degraded through environmentally friendly methods. The non-thermal plasma-based advanced oxidation process (AOP) is one of the most widely studied and best developed processes owing to its simple structure and ease of operation. In this study, a plasma-based AOP was stably generated using submerged multi-hole dielectric barrier discharge (DBD) and applied to relatively high concentration of TBP solution. A submerged DBD plasma system was designed to directly interact with water, thereby producing reactive oxygen species (ROS) and functioning as a powerful oxidizer. Additionally, UV, O₃, and H₂O₂ are generated by the developed plasma system without using any other additives to produce OH radicals for degrading organic pollutants; therefore, this system circumvents the use of complex and advanced oxidation processes. The electrical properties and concentrations of the active species were analyzed to establish optimal plasma operating conditions for degrading TBP solution. The results were analyzed by measuring the total organic carbon (TOC) and changes in solution properties. Based on these results, a degradation mechanism of TBP solution is proposed. After 50 min of plasma treatment, the concentration of TOC was gradually decreased. Consequently, we found that plasma-based AOP using submerged multi-hole DBD has advantages as an alternative technology for degrading organic pollutants such as TBP solution.

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Keywords: Submerged multi-hole dielectric barrier discharge, Plasma-based advanced oxidation process, Tributyl phosphate, Organic pollutant, OH radical

Development of a Treatment System for Decommissioning Radwaste

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During the operation or decommission of nuclear facilities, a large amount of dry active waste and cable waste with various shape and material is generated. Most of these wastes have almost no radioactive contamination and can be disposed of by incineration, landfill, recycling, etc. under clearance regulation. For clearance of radioactive waste, it is necessary to verify the characteristics of radiological contamination and prove that it meets the criteria for clearance regulation. According to the domestic clearance regulation, if it is difficult to measure radioactivity of wastes due to their surface condition using direct or indirect measurement methods, representative samples should be collected and analyzed for radioactivity. When sampling, it is desirable to collect samples of about 1 kg that can represent waste contamination per 200 kg or per 1 m², and the homogeneity of the samples also should be demonstrated. However, in the case of dry active wastes, it is very difficult to prove the homogeneity of the samples because of surface shapes and conditions of the wastes. In particular, considering cable waste generated during the decommission, it is hardly capable to prove the representativeness of the sample, even though the inner shell of the covering material and the copper wire are almost uncontaminated. In this study, we show the development of a treatment system that makes it easy to prove the representativeness of samples when disposing of dry active waste or cable waste generated in nuclear facilities. The treatment device is designed in such a way that it has different storage unit and cutting unit suitable for the material characteristics of each waste type (soft, hard and cable), and therefore optimizes the efficiency of the shredding or cutting process. In addition, it is expected that the work efficiency in the radioactive treatment site with a narrow area can also be improved by providing a moving part on the device.

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Keywords: Radioactive waste, Treatment device, Dry active waste, Cable waste

Preliminary Approach to Optimization of Scenarios for Predisposal Management of Radioactive Waste Using System Dynamics Technique

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Waste that contains or is contaminated with radionuclides arises from a number of activities involving the use of radioactive material. Such activities include the operation and decommissioning of nuclear facilities; the use of radionuclides in medicine, industry, agriculture, research and education. Radioactive waste must be safely disposed in a radioactive waste repository for the protection of public health and the environment. In order to safely dispose of radioactive waste in a repository, it is important to derive an optimal predisposal management scenario because radioactive waste must be processed (i.e. processing (pretreatment, treatment and conditioning), storage and transport) for satisfying waste acceptance criteria (WAC). Optimal scenario of predisposal management of radioactive waste is derived for considering the balancing of exposures of workers and/or those of members of the public, the short term and long term risk implications of different waste management strategies, the technological options available and the costs. However, existing studies for deriving the optimal scenario of predisposal management of radioactive waste have evaluated only the radiation dose of workers and public within given scenarios using fixed value, or have derived optimal single process (i.e. decontamination) of predisposal management using Multi-Attribute Decision Making (MADM) methodology. In this study, optimal predisposal management scenario is derived by evaluating exposures of workers using system dynamics (SD) technique. Radiation dose assessment SD model was modeled using VENSIM® code developed by VENTANA systems Inc.. SD Model has the advantage of being able to respond flexibly when decision makers want to change input data and it has the advantage of being able to track dynamically changing phenomena and visually confirm interdependence. After that, based on the SD model derived from this study, evaluations of exposures of public, cost, and technicality will be added to be utilized when establishing an optimal scenario of predisposal management of radioactive waste considering multi attribute.

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Keywords: System dynamics, Vensim®, Radioactive waste, Predisposal management, Optimisation

Construction of Measures to Ensure Mobile Stability of a Movable Radionuclide Analysis System

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As the plan for the nuclear dismantlement due to the permanent shutdown of Kori-1 and Wolsong-1 nuclear power plants has been concretized, a “movable radionuclide analysis system” is being developed that can quickly and accurately analyze large amounts of radioactive waste generated on the sites during dismantling. This system has various advantages from the perspective of strict regulations on the radioactive waste movement and social acceptability, such as preventing unexpected accidents while moving on the national highway or expressway, reducing various documents and immediate response to dismantling plans. Currently the system is being developed to be equipped with previously developed sample pretreatment and radioactivity measuring equipment and automated volatile and nonvolatile nuclide separation equipments, but to ensure mobile stability, it needs to analyze factors and establish stability standards. In the KS Q ISO/IEC 17025:2017 standard, the requirements for “facilities and environmental conditions” are a very important factor in building reliability for consumers as part of the quality guarantee for this facility. In order to meet the requirements, the technical standards of various test equipment to be installed in this facility were investigated. The physical, chemical, and radiological hazards that could affect the safety of the equipment and workers in the process of moving the equipment between nuclear power plants or between nuclear dismantling sites were derived from vibrations, rapid changes in temperature and humidity, and the spread of contamination from radioactive waste samples. Therefore, the scope of application of the law, which is the basis for securing stability during movement, was classified into two situations: movement from facility manufacturer to installation site (non-contaminated) and movement from primary to secondary use (contaminated). And in order to investigate the Nuclear Safety Act, enforcement ordinances, and radiation safety management, and to establish standards for packaging and transportation of radioactive materials, the results of transportation tests and transport details were compared and analyzed. Finally, the air suspension systems and the automatic temperature and humidity control devices were analyzed to establish standards for securing stability against the vibration and the sharp changes in the temperature and humidity, and countermeasures such as accident measures in accordance with the Enforcement Decree of the Nuclear Safety Act were also investigated.

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Keywords: Movable radionuclide analysis facility, Mobile stability

Development of Plasma Torch Waste Feeding System

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Plasma torch melting technology can pyrolyze and melt waste with high-temperature heat (about 1,600°C) using electric arc phenomena such as lightning. Waste that may be treated in a plasma torch melting facility is injected in solid (combustible, non-combustible) and liquid form depending on facility capacity. The 200 L drum type, screw supply type, and nozzle type liquid injection device are applied to MW plasma facilities, and the push rod type and screw supply type are applied to small-capacity plasma facilities. In consideration of the characteristics of radioactive waste generated from operating and dismantling nuclear power plants, a waste input device suitable for plasma torch facilities was developed and verified through tests. In the future, facility soundness will be confirmed through long-term performance tests, and stability will be secured through continuous improvement.

Keywords: Plasma, Waste, 200 L drum, Torch

A Study on the Treatment of Insulation Using Plasma Torch Melting Technology

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In nuclear power plants, insulation is used to protect equipment and block heat. Insulation materials include asbestos, glass fiber, calcium silicate, etc. Various types and materials are used. This study aims to ensure volume reduction and disposal safety by applying plasma torch melting technology to insulation generated at operating and dismantling nuclear power plants. After the evaluation of characteristics by securing thermal insulation materials or similar materials in use at the operational and dismantling nuclear power plant. It is planned to perform pyrolysis and melting tests using the MW plasma torch melting facility owned by KHNP CRI Before the plasma test, check the thermal decomposition and melting characteristics (fluidity, etc.) of the insulation in a 1,600°C high-temperature furnace. The insulation is stored in a 200 L drum and injected into a plasma facility, and the drum and the insulation are to be pyrolyzed and melted by the high temperature inside the plasma torch melting furnace. Through this test, thermal decomposition and melting of the insulation, solidification/stabilization method, maximum throughput, and exhaust characteristics are confirmed at a high temperature (1,600°C) of the plasma torch. Through this study, it is expected that the stable treatment and disposal of insulation generated from operating and dismantling nuclear power plants will be possible.

Keywords: Plasma, Insulation, Melting, Torch

Development of 200 L Drum Treatment Technology Using Plasma Torch Melting Facility

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Currently, KHNP has 24 operating nuclear power plant units with a total combined capacity of about 23 GWe and two units are under construction. However, permanent stop of Kori unit 1 nuclear power plant was decided in 2017. Accordingly, interest in how to dispose of waste stored inside a permanently stopped nuclear power plant and waste generated as decommissioning process is increasing. KHNP CRI is conducting research on the advancement of plasma torch melting facilities for waste treatment generated during the plant decommissioning and operation period. Plasma torch melting facility is composed of various equipment such as a melting furnace (Melting chamber, Pyrolysis chamber), a torch, an exhaust system facility, a waste supply device, and other equipment. In demonstration test, concrete waste was put in a 200 L drum to check whether it can be pyrolyzed using a plasma torch melting facility. Reproducibility for waste treatment in the form of a 200 L drum and discharge of molten slag could be confirmed, the amount of concrete waste in 200 L Drum that could be treated according to power of plasma torch was confirmed. This demonstration test confirmed the field applicability and stability of plasma torch melting facility, and improved expectations for long-term operation.

Keywords: Plasma torch melting facility, Decommissioning waste, 200 L Drum, Molten slag

Arrangement of Subsequent Disposal Facilities in the Complex Disposal Facility

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Currently, the Gyeongju disposal facility is planned to be operated as a complex disposal facility with three types: cave disposal, surface disposal, and landfill disposal. Approximate method and arrangement have been decided up to the 1st, 2nd, and 3rd stage disposal facilities, but the optimal method for the arrangement of the entire complex disposal facility has not been established. When establishing the subsequent disposal facility arrangement plan, the generation prospect and disposal capacity setting plan for each level of radioactive waste was established, and the disposal capacity of the subsequent disposal facility, the disposal facility method, etc. were reviewed and reflected. Among the items for deriving an efficient management plan, KEPCO E&C is going to first derive a site arrangement plan for each disposal facility of 800,000 drums of radioactive waste, and has drawn up a plan for each scenario through collaboration with other organizations. When establishing the subsequent disposal facility arrangement plan, the generation prospect and disposal capacity setting plan for each level of radioactive waste was established, and the disposal capacity of the subsequent disposal facility, the disposal facility method, etc. were reviewed and reflected.

Keywords: Complex disposal facility, Arrangement plan

Measurement of Cement Solidified Radioactive Waste According to the Compressive Test Condition

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Low-and intermediate level waste (LILW) should be solidified and satisfy the waste acceptance criteria (WAC) to be disposed of in the LILW repository. The LILW should be uniformly solidified and should maintain its structural stability under the expected condition according to the WAC. Compressive strength of cement solidified waste should satisfy at least 3.44 MPa to be disposed of in the repository. In addition, its compressive strength should satisfy at least 3.44 MPa after the irradiation, immersion and leaching test. The compressive strength test and dimension of test specimen differ according to countries. However, measured compressive strength of solidified waste is affected by geometry of specimen and test condition. Diameter, ratio between diameter and height, and porosity are one of factors that affect to the compressive strength of cement solidified waste. Generally, specimen with larger diameter shows higher value of measured compressive strength. The ratio of height and diameter shows similar tendency to the diameter while larger porosity generally lowers the compressive strength. In other hands, higher compressive strength is expected when the loading rate is higher during the compressive strength test. U.S. is applying loading rate from ASTM C39 (0.25 ± 0.05 MPa) for the compressive strength test while Korea is applying loading rate from KS F 2405 ($0.6 \text{ MPa} \cdot \text{s}^{-1}$). France applies loading rate following FT-02-010 ($0.5 \text{ MPa} \cdot \text{s}^{-1}$) for cement solidified waste. As the measured compressive strength increases when the loading rate increases, the effect of loading rate to the compressive strength of cement solidified waste should be assessed by quantification and consider its effect on the sight of regulation. In this study, the effect of geometric parameters of specimen and test condition to the compressive strength are checked by manufacturing specimen by solidifying mock sludge waste with cement. To prevent increasing amount of secondary waste, effects of ratio of height and diameter and porosity to the compressive strength are checked while diameter value is fixed. For loading rate, loading rate from ASTM C39 and KS F 2405 were compared. Existence of significant variance of measured compressive strengths of cement solidified waste are check by performing statistical analysis. Finally, by analyzing the relationship between test condition and measured compressive strength, the test method that measures the compressive strength conservatively is aimed to be derived.

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Keywords: Compressive strength, Loading rate, Porosity, Repository, Solidified waste

Correlation of Radiological Characteristics of Solid Radioactive Waste Generated From US PWRs During Operation and After Permanent Shutdown

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Kori and Wolsong unit 1 were permanently shutdown in 2017 and 2019, respectively. During the decommissioning of a nuclear power plant, various types and levels of decommissioning waste will be generated sporadically in many areas in a relatively short period of time, so safe management of decommissioning waste is expected to emerge as a very important issue in the future. Since Korea has no experience in decommissioning nuclear power plants, radionuclides added by abnormal routes or errors in data can be identified through the list of expected nuclides and radioactivity data during decommissioning by analyzing cases of overseas nuclear power plants decommissioning. Therefore, it is expected that safety information of nuclear power plants in the United States (i.e. all information related to safety, such as radioactive waste characteristics and accident or decommissioning information at nuclear power plants) can be utilized when decommissioning Korea nuclear power plants.

Therefore, in this study, the characteristics of solid radioactive waste were analyzed by collecting solid radioactive waste data during operation and after permanent shutdown of nine PWR nuclear power plants in the United States, and the correlation between the characteristics data of solid radioactive waste was analyzed. However, in the case of Korea, only data from the United States were analyzed because there was no data for each radionuclide that were disclosed when disposing of radioactive waste in LILW repository and there was no nuclear power plant that had been decommissioned. Correlation analysis of solid radioactive waste was performed by linking radioactivity of radionuclides, volume of waste, and total radioactivity data based on decommissioning work and accident data after permanent shutdown or during operation. The correlation analysis of total radioactivity, volume, and radioactivity of each nuclide of solid radioactive waste during operation and after permanent shutdown was performed using XLSTAT, an Excel add-in software, for carrying out Mann-Kendall Test and estimating Sen's slope. Trends during operation and after permanent shutdown were compared and the effects of specific events or tasks were analyzed. This study is expected to be utilized as basic data related to safety management of decommissioning Korea nuclear power plants in future.

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Keywords: Correlation analysis, Decommissioning, Mann-Kendall test, Radioactive waste, Sen's slope

Effect of Sulfate Radical on Advanced Oxidation Process for Treatment of Liquid Scintillation Waste

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Liquid scintillation cocktail is liquid waste, which consists of an organic solvent, scintillator, surfactant, and radionuclide. Large volumes of liquid scintillation waste are generated each year, and both the organic compound and radionuclide content can negatively affect on the health and the environment. Therefore, the liquid scintillation waste should be treated in an appropriate way. In this study, to facilitate the treatment of liquid scintillation waste, the sulfate-radical advanced oxidation process (SR-AOP) was performed for the mineralization of liquid scintillator waste. In SR-AOP, highly reactive sulfate radicals, which react more selectively and efficiently with organic compounds, are produced in situ by cleaving the peroxide bond in the persulfate molecule. For the experiment, 100 times diluted ULTIMA GOLD-LLT (initial TOC=699,800 ppm) was used as a liquid scintillation waste. The TOC removal efficiency of liquid scintillation waste by the OXONE (potassium peroxydisulfate, PMS, $2\text{KHSO}_5 + \text{KHSO}_4 + \text{K}_2\text{SO}_4$) and sodium persulfate (PS) with varying dosages (4–12 mM) was tested, and the effects of Co^{2+} and Cu^{2+} catalysts were compared at a range of pHs (3, 7, and 9). The experimental results demonstrated that 91% TOC removal of ULTIMA GOLD-LLT could be achieved for SR-AOP at initial pH=9, Co^{2+} =1.2 mM (catalyst), PMS=4.8 mM (oxidant) for 60 min reaction. Compared to traditional Fenton AOP which is effective only at low pH, PMS based SR-AOP with Co^{2+} catalyst is effective at wide range of pHs and less dependent on the treatment efficiency of the operational pH. Therefore, it can be useful for the mineralization of liquid scintillation waste which is difficult to treat with a general treatment method due to the mixture of various organic compounds.

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Keywords: LSC cocktail, Advance oxidation process, Potassium peroxydisulfate, Sulfate radical

Application of GASPARD Code for Regulatory Clearance

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According to the Atomic Energy Act of Korea, radioactive waste can be cleared when it meets the criteria, less than $10 \text{ uSv} \cdot \text{y}^{-1}$ for individual dose and $1 \text{ person} \cdot \text{Sv} \cdot \text{y}^{-1}$ for collective dose. Consequently, it is necessary to evaluate radiation dose to get permission for regulatory clearance from the regulatory body of Korea. Several computational programs can be used for dose calculation depending on disposal methods such as landfill, incineration, and recycling. As for incineration, the effects of radionuclide emitted during combusting radwaste have to be considered to figure out exposure dose. In this study, GASPARD code is described to assess exposure dose from effluents released to the atmosphere during incinerating combustible radioactive wastes for regulatory clearance.

GASPARD is the code programmed by Radiation Safety Information Computational Center at Oak Ridge National Laboratory for computing annual dose due to radioactive effluents released from a nuclear power plant to the atmosphere during routine operation. The calculating methods of the code are based on the mathematical model of U.S. NRC regulatory guide 1.109, about beta and gamma radiation from noble gas in semi-infinite plume, radioiodine, and particulates. GASPARD evaluates both individual dose and population dose. The considering pathways are composed of external exposure by plume and ground deposition of effluents, and internal exposure as a result of inhalation and food ingestion.

Since the calculation model of GASPARD requires various variables about the radionuclide and disposal site, the accuracy of the results is decided by inputted values. The program contains the default values to parameters such as the humidity, fraction of deposition, and storage time of foods. However, to get permission, it is important to use the appropriate data representing the condition of the combustion scenario as substitutes for the default since the values are localized to the country where the code was developed. Therefore, dose assessment by GASPARD code can be applied for regulatory clearance by incineration, when reliable values depending on the disposal plan inputted.

Keywords: GASPARD, Regulatory clearance, Dose assessment, Airborne effluents

A Study on the Performance Improvement and Analysis Method of the LILW Disposal Container Cover

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Radioactive waste disposal facility in Korea, radioactive waste packaged in 200 L drums is placed in a concrete disposal container and disposed of at an underground silo type (cave) disposal facility. At this time, the disposal container cover is seated on the top of the disposal container, and if the disposal container and the cover are not completely combined, the container cover is raised up from the top of the disposal container, so safety problems may occur when stacking the disposal container. Therefore, various methods exist to secure a margin for the pure height inside the disposal container. The disposal container cover only covers the upper surface of the container to shield radiation, and structural performance is not required. Therefore, the method of processing the cover, such as a method of making the cover of the disposal container thin, is the easiest method to apply. In this study, a method to reduce the thickness of the cover of a concrete disposal container was devised, and structural performance under usability conditions such as lifting and seating was analyzed. In addition, the disposal container cover has a reinforced concrete form in which dissimilar materials (concrete and steel) are combined, an integrated analysis was performed to secure the reliability of the analysis results for this, and the analysis results were described. It was found that the proposed disposal container cover structure can improve usability by reducing the stress concentration phenomenon.

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Keywords: LILW, Disposal container, Stress concentration, FEM

Evaluation of Leaching Behavior of Mixed Waste in the Simulated Radioactive Waste Disposal Facility Environment

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In nuclear power plants and nuclear facilities, radioactive waste containing hazardous substances (Mixed waste) is continuously generated due to research such as radiochemical study and nuclide analysis. In addition, radioactive waste including heavy metals and asbestos is generated during the dismantling process of nuclear power plants. Mixed wastes have both radiation hazards and chemical hazards, and there's a possibility of synergistic effects generation. However, in most countries except the United States, there are no regulatory standards for the chemical hazards of mixed waste. The regulations applicable to mixed waste in Korea include the Nuclear Safety Act and the Waste Management Act. The Nuclear Safety Act prohibits the acceptance of hazardous radioactive waste in disposal facilities, but there is no definition or characteristic identification procedure for "hazardous." The Waste Management Act also does not state the regulation for radioactive waste. In the Gyeongju disposal facility in Korea, the leachate in the disposal facility is expected to be a groundwater saturated with concrete and is expected to be irradiated by radioactive waste. On the other hands, most of the non-radioactive waste landfill facilities are built on the surface, and the leachate is expected to be rainwater that reacts with the soil. Due to the differences in leaching environments, there's a potential to overestimate or underestimate the leaching properties of hazardous substances if the standard leaching test is applied. To show for this, a leaching test simulating disposal facility's environment were applied to sample waste containing heavy metals. The leaching solution was groundwater collected from the area near the Gyeongju disposal facility, which is then saturated with concrete and adjusted to pH 12.5. In addition, gamma-ray irradiation was conducted during the leaching test to observe changes in the leaching behavior of heavy metals in the actual radioactive waste disposal environment. As a result, lead showed significantly increased leaching compared to the standard test method, and cadmium was not detected in all experimental conditions except heavy irradiation. This study suggested that regulations on the hazardous of mixed waste should be settled, which should be established in sufficient consideration of the types and characteristics of substances contained in the waste.

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Keywords: Mixed waste, Hazardous, Heavy metal, Leaching test

The Solidification of Radioactive Sludge Powder Based on Metallic Oxide and Ferro Frit Powder by Hot Isostatic Pressing

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The mixing powder of vitrification material and metallic oxide sludge was solidified by hot isostatic press method and was tested to check whether the solidified waste disposal acceptance criteria were met or not. From various contaminated tank in nuclear power plants, and other nuclear energy facilities, radioactive sludge based on metallic oxide can be generated. The most of tank consist of stainless steel can be oxidated by the long-term exposure on oxygen and moisture, and then can be made sludge layer based on metallic oxide on the inner wall of contaminated tank. Radioactive sludge waste should be solidified and disposed. Melting and hardening is the most basic method for solidification. The melting points of metallic oxide of stainless steel as Fe_3O_4 , NiO , Cr_2O_3 are 1597, 1955, 2435, respectively. Those are very high temperature. To melt these metallic oxides, a furnace capable of raising the temperature to a very high temperature is required, which requires a lot of thermal energy, which may lead to an increase in disposal cost. Therefore, it is necessary to lower the melting point and solidify non-melted metallic oxide powder by adding vitrifying material powder as Na_2O , SiO_2 , B_2O_3 . The more vitrification material is added, the easier it is to solidify the sludge based on metallic powder at a low temperature, but there is a problem in that the total waste volume increases due to the addition of vitrification material. In this study, the mixing ratio and temperature conditions that can fix the sludge while adding a minimum amount of vitrification material will be confirmed. Mixing ratio conditions of the vitrification material and sludge powder are 10:90, 15:85, 20:80, 25:75. To fix the metallic oxide sludge by melting only the vitrification material without completely melting the metallic oxide, compression by external pressure is required. Therefore, the HIP (Hot Isostatic Pressing) method was used to solidify the metallic oxide sludge by simultaneously heating and pressurizing it. Because the softening points of most of vitrification based on Na_2O , SiO_2 , B_2O_3 are ranged from 800 to 1000, temperature conditions are 800, 900, 1000. Since the compressive strength for disposing of the solidified materials was 3.4 MPa, the maximum pressure condition was set to 5000 psi (about 34 MPa), which is 10 times 3.4 MPa. And optimal mixing ratio, temperature, pressure conditions that meet the solidified waste disposal acceptance criteria will be found.

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Keywords: Radioactive sludge, HIP, Vitrification, Leachability index

Development of C-14 Treatment Facility for C-14 Treatment in PHWR

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The permanent shutdown of Wolsong 1, PHWR (Pressurized Heavy Water Reactor) was decided. Accordingly, there is need for C-14 treatment technology to spent resin generated by PHWR in classified Medium Level Radioactive Waste by C-14 specific activity. However, spent resin by PHWR is mixed and stored with activated carbon and zeolite (mixture), not a single storage, and separation from the mixture must be carried out in advance for C-14 treatment in the spent resin. This study developed a C-14 treatment facility that combined with the technology of separating spent resin from spent resin mixture by PHWR NPP and the technology of C-14 treatment for disposal. The C-14 treatment facility consists of spent resin separation (Part 1) and treatment of separated spent resin. (Part 2) Part 1 is applied with a process of separating the mixed and stored spent resin from the spent resin mixture by applying a drum screen method. In the case of Part 2, spent resin treatment process for desorbing and collecting C-14 nuclides in the separated spent resin using microwave reactor was applied. Except for the adsorbent used to collect C-14 detached in the process of separating and treating spent resin, no additional material is introduced into the facility, and thus secondary waste is significantly reduced. In addition, pollution prevention banks at the bottom of the facility and a sealed automated circulation system were applied to prevent unexpected leakage and diffusion of radioactive materials and ensure stability of workers. Currently, the C-14 treatment facility has been verified for spent resin separation and spent resin treatment using simulated spent resin mixture, and the facility will be demonstrated and verified for field applicability. According to derived results, it is believed that it will be possible to apply the C-14 treatment facility when decommissioning of PHWR.

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Keywords: PHWR, Spent resin, C-14, C-14 treatment facility, Field applicability

Introduction of Test Devices for Radwaste Characterization Specimen

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In this introduction, test devices for radwaste characterization specimen was developed and utilized. In order to permanently dispose of solidified radwastes, not only radioactive characterization but also physical & chemical characterization shall be performed to assess compliance with the waste acceptance criteria.

Waste acceptance criteria can be made up measurement of free standing water, compressive strength test, thermal cycling test, radiation resistance test, leaching test, immersion test.

Approximately, the equipment for each test is sorted out five types. equipment for making characterization specimen, equipment for compressive strength test, equipment for thermal cycling test, equipment for radiation resistance test, equipment for Immersion test and leaching test.

Equipment for making characterization specimen is operated the dry process. The equipment of two types: one (sampling device) that cores solidified radioactive waste in a drum, and the other (cutting machine) that properly cuts the coring samples. Sampling device is not used in industry, so it is specially manufactured, cutting machine is using ready-made products. In addition, devices for compressive strength test and thermal cycling test are using ready-made products. Facility for Radiation resistance test is located in Jeong-eup. For the efficient test, a table was manufactured in the columnar form like the specimen. Finally, devices for immersion test and leaching test are so transformed that contact all surfaces of the specimen with the liquid.

Keywords: Radioactive waste, Characterization, Waste acceptance criteria

Feasibility Study of Synthesizing Graphene Quantum Dots From the Spent Resin in a Nuclear Power Plant

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The feasibility study of synthesizing graphene quantum dots from spent resin, which is used in nuclear power plants to purify the liquid radioactive waste, was conducted. Owing to radiation safety and regulatory issues, an uncontaminated ion-exchange resin, IRN150 H/OH, prior to its use in a nuclear power plant, was used as the material of experiment on synthesis of graphene quantum dots. Since the major radionuclides in spent resin are treated by thermal decomposition, prior to conducting the experiment, carbonization of ion-exchange resin was performed. The experiment on synthesis of graphene quantum dots was conducted according to the general hydrothermal/solvothermal synthesis method as follows. The carbonized ion-exchange resin was added to a solution, which is a mixture of sulfuric acid and nitric acid in ratio of 3:1, and graphene quantum dots were synthesized at 115°C for 48 hours. After synthesizing, procedure, such as purifying, filtering, evaporating were conducted to remove residual acid from the graphene quantum dots. After freeze-drying which is the last procedure, the graphene quantum dots were obtained. The obtained graphene quantum dots were characterized using atomic force microscopy (AFM), Fourier-transform infrared (FT-IR) spectroscopy and Raman spectroscopy. The AFM image demonstrates the topographic morphology of obtained graphene quantum dots, the heights of which range from 0.4 to 3 nm, corresponding to 1–4 graphene layers, and the step height is approximately 2–2.5 nm. Using FT-IR, the functional groups in obtained graphene quantum dots were detected. The stretching vibrations of hydroxyl group at 3,420 cm^{-1} , carboxylic acid (C=O) at 1,751 cm^{-1} , C-OH at 1,445 cm^{-1} , and C-O at 1,054 cm^{-1} . The identified functional groups of obtained graphene quantum dots matched the functional groups which are present if it is a graphene quantum dot. In Raman spectrum, the D and G peaks, which are the characteristics of graphene quantum dots, were detected at wavenumbers of 1,380 cm^{-1} and 1,580 cm^{-1} , respectively. Thus, it was verified that the graphene quantum dots could be successfully synthesized from the ion-exchange resin.

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Keywords: Radioactive waste, Spent resin, Recycle, Graphene quantum dots

Development of Small Air Purifying Equipment Using Used Mask Filter

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The background of the development is to contribute to the reduction of radioactive waste, recycling of resources and effectively purifying the air in the workplace. Ultimately, it affects the reduction of internal exposure of workers. According to the standard procedure of KHNP, 「Use and Management of Respiratory Protection Equipment」, the expiration date of mask filter is indicated by the manufacturer before opening. It is 1 year from the date of first combination after opening. We have developed an air purifying equipment that can recycle and reuse expired mask filter waste in nuclear power plant. In order to confirm the performance, we observed air pollution level by operation time. The location was measured at 3 locations including the decontamination product warehouse in NPP, and the size of the measurement locations were 53 m³, 150 m³, 180 m³, and 900 m³. As a result of measurement, significant air purification effect was found in 53 m³ and 150 m³. Decontamination effect of 80% was shown after 1 hour of operation, and 20% of decontamination effect was shown gently for 3 hours thereafter. On the other hand, there was no significant decontamination effect in the 180 m³ and 900 m³ spaces. Significant results were derived by statistical methods. Statistical procedure involves the collection of data leading to test of the relationship between two statistical data sets, or a data set and synthetic data drawn from an idealized model. The basic composition and product characteristics was as follows: Blower, filter fixing unit, Air purifier outlet round shape, Differential pressure gauge, inverter (200 V, 3 π , 200 W). The developed product weigh is 25 kg. This is lighter than the existing product weighing 100 kg. It is judged that it is suitable for convenient use. Because the area where the major air pollution level occurs is isolated by a room in NPP.

This developed product has a greater significance in that it recycles radioactive waste within the radiation management area rather than air purification efficiency.

Keywords: Air purification, Radioactive waste, Mask filter

A Study on the Impact of Animal Intrusion Into Near-Surface Disposal Facility for LLW Radioactive Waste

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Near-surface disposal facility is more susceptible to intrusion than underground repository, resulting in more possible pathways for contaminant release. Alike human intrusion, animals (e.g. Ants, Moles, etc.) could intrude into the disposal site to excavate burrows, which could cause direct release of contaminants to biosphere. In this paper, animal intrusion is demonstrated using GoldSim's commercial contaminant transport module and impact on the integrity of the near-surface disposal facility is evaluated in terms of fractional release rate of the contaminants.

In this study, the near-surface disposal facility is modelled with a single concrete vault to contain radionuclide according to LLW concentration limit stated in NSSC notice No.2020-6. The release of contaminants is modelled to occur directly after the institutional control period, and the contaminants are mostly transported from the concrete vault to cover layers via diffusion.

To produce mathematical model of the release of the contaminants due to animal intrusion, firstly, the fraction of burrow volume for each cover layer is calculated separately for each animal species, based on their maximum possible intrusion depth. In this study, fractions of burrow volume for ants and moles are calculated based on their maximum possible intrusion depths, where for ants is 2–3 m, and for moles is 0.1–0.135 m. Then, assuming that the contaminants are distributed homogeneously throughout each cover layers by diffusion, fraction of contaminants transported into the uppermost layer via excavation of the burrow is calculated for each layer based on burrow volume, and fraction of contaminants removed from the uppermost layer to the layers below via collapse of the burrow is also calculated based on the burrow volume. Lastly, the net transportation of contaminants into and out of the burrow via excavation and collapse, respectively, is calculated and demonstrated using direct transfer rate function of the GoldSim.

Based on the simulated result, the maximum mass flux is too minor to cause a meaningful impact on the safety. The peak mass flux of the most sensitive radionuclide, I-129, is witnessed at around year 1,470, with a flux value of $5.36 \times 10^{-6} \text{ g} \cdot \text{yr}^{-1}$. This minor release of the contaminants could be due to cover layers being much thicker than the maximum possible intrusion depth of the animals, preventing the animal intrusion into the deeper layers of higher radionuclide concentration. In future, this study can be used to provide a guidance and fundamental data for scenario development and safety evaluation of the near-surface disposal facility.

Keywords: Post-Closure safety evaluation, Near-Surface disposal facility, Animal intrusion, GoldSim, Contaminant transport

A Study on the Factors Required to Be Considered for Safety Assessment of Cellulose Degradation

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Recently, concern regarding disposal of cellulosic material is growing as cellulose is known to produce complexing agent, isosaccharinic acid (ISA), upon degradation. ISA could enhance mobility of some radionuclides, thus increasing the amount of radionuclide released into the environment. Thus, evaluation on the possible impact of the cellulose degradation would be an important aspect in safety evaluation. In this paper, safety assessments conducted in Sweden and UK are studied, and the factors required to be considered for appropriate safety assessment of cellulose is analyzed.

SKB (Sweden) conducted safety assessment of cellulose degradation as a part of long-term safety assessment of SFR. SKB determined that ISA would impact sorption of trivalent and tetravalent radionuclides (Eu, Am, Th, Np, Pa, Pu, U, Tc, Zr and Nb) at concentration higher than 10^{-4} – 10^{-3} M, and impact sorption of divalent radionuclides (Ni, Co, Fe, Be and Pb) at concentration higher than 10^{-2} M.

Then, SKB conservatively set the upper limit of ISA concentration to be 10^{-4} M and conducted cellulose degradation evaluation on each waste package type, considering the expected disposal environment of SFR. Based on the calculated results, some of the waste packages showed concentration of ISA to be higher than 10^{-4} M, so SKB conservatively developed waste acceptance criteria to prevent ISA being produced to an extent of affecting the safety of the repository.

SKB conducted safety assessment only for the repositories with pH above 12.5 and excluded 1BLA from the safety assessment as the expected pH of 1BLA is around 12, which is insufficient for cellulose to degrade. However, SKB set disposal limit for 1BLA as well, to minimize potential impact in future.

Serco (UK) conducted safety assessment of cellulose degradation for the conceptual repository, which is a concrete vault with cementitious backfill. Serco estimated that the pH of repository would maintain around 12.4. Serco conservatively assumed that the pH would be sufficient for cellulose degradation to occur partially, and suggested application of appropriate degradation ratio for safety assessment of cellulose degradation.

To conduct appropriate safety assessment of cellulose degradation, an appropriate ISA concentration limit based on radionuclide inventory list, and an appropriate cellulose degradation ratio based on the pH of disposal environment should be determined. As for guidance, below pH 12.5, cellulose degradation is not expected, and between pH 12.5–13, partial cellulose degradation is expected. In future, this study could be used as fundamental data to evaluate safety of the repository.

Keywords: Cellulose Degradation, ISA, Disposal Environment, Safety Assessment

Development of Slag Composition of Concrete Waste for Plasma Torch Melting System

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Plasma melting technology has been considered as promising technology for treatment of radioactive wastes. According to the IAEA TECDOC-1527 report (2006), the technology has an advantage that it can treat regardless of waste types which is both combustible and non-combustible wastes. In particular, it is expected that a large amount of concrete, a representative non-combustible wastes, will be generated during the operation and dismantling of nuclear power plants. In order to treat the concrete waste in plasma torch melting system, various factors could be considered like the slag of electric conductivity, viscosity and melting temperature. Above all, as a critical factor, the viscosity of the melt is very important to easily discharge the melt. The viscosity of slag ($\text{SiO}_2\text{-CaO-Al}_2\text{O}_3$ system) can be lowered by adding a basic oxide such as CaO , Na_2O , MgO and MnO . The basic oxides are donors of oxygen ions. These oxides are called network breakers, because they destroy the network of SiO_2 by reacting with it. In this study, the slag composition of the concrete waste was developed to apply the plasma torch melting. Also, demonstration test was performed with the developed slag composition and 100 kW plasma torch melting system.

Keywords: Concrete, Plasma torch melting system, Slag composition, Demonstration test

Evaluation of Applicability of 100 kW Plasma Torch Melting System for Combustible Waste

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In KHNP CRI, the 100 kW PTM (plasma torch melting) system was designed for the treatment and disposal technology of various radioactive wastes including the metal, concrete, liquid waste and insulator. The facility consists of melting chamber, thermal decomposition chamber, waste feeding system and off-gas treatment system. In this study, to evaluate the applicability of the PTM system, demonstration test was conducted using the radiation hazmat suit as combustible waste. The plasma melting chamber is pre-heated by 2nd combustion device and plasma torch for 5 hours. The temperature inside the plasma melting chamber is approximately 1,600°C. The combustible waste was put into the melting chamber by the pusher feeding device with the throughput of maximum 50 kg/hour. During the test, the power of plasma torch is 60–96 kW on the transferred mod. It was evaluated in terms of long-term integrity of PTM system on operation according to the waste throughput ratio.

Keywords: 100 kW PTM system, Combustible waste, Demonstration test, Throughput

Review of Candidate Technologies for Thermal Treatment of Sludge Radwaste Generated During Fukushima Contaminated Water Treatment

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After the Fukushima accident in 2011, a huge amount of radioactively contaminated water is being generated by cooling the melted fuel of units 1, 2 and 3. Most of contaminated water is seawater and underwater containing not only salt elements but also nuclear fission products with radioactivity. To treat the contaminated water, Cs/Sr removal facilities such as KURION and SARRY are being operated by TEPCO. Additionally, three ALPS facilities are on operation to meet the regularity standards for discharge to the sea. However, massive secondary wastes such as Zeolite, sludge and adsorbent is being generated by these facilities for liquid water treatment. The secondary wastes containing various radionuclide with Cs and Sr is difficult to store due to highly radioactive concentration and corrosive properties. In Japan, a variety of technologies such as GeoMelt vitrification, In-Can vitrification and CCIM vitrification is considered as a promising solution. In this study, they were reviewed, and the advantage and disadvantage of each technology were evaluated as the candidate technologies for thermal treatment of sludge radwaste.

Keywords: Fukushima, Contaminated water, Secondary waste, Thermal treatment, Vitrification

A Study on the Characterization for Polymer Waste Form of Fine Contaminated Soil Generated After Soil Washing

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Radioactive materials emitted from nuclear accident or decommissioning cause soil contamination over wide areas. In the event of such a wide area of contaminated soil, decontamination is inevitable for residents to reside and reuse as industrial land. There are many ways to decontaminate these contaminated soils, but in urgent situations, the soil washing, which has a short process period and relatively high decontamination efficiency, is considered the most suitable. However, the soil washing process of removing fine soil and cesium by using washing liquid as water and adding a flocculating agent (J-AF) generates slurry/sludge-type secondary waste (Cs-contaminated soil + flocculating agent). Since this form of sludge contaminants cannot be disposed, solidification is needed using an appropriate solidification agent to treat wastes for disposal. Therefore, this study devised a treatment method of contaminated fine soils occurring after the soil washing process. This investigation prepared the simulated wastes of contaminated fine soils generated after the soil washing, and pelletized the samples using a roll compactor under the optimum operating conditions. The optimum conditions of the device were determined in the pre-test. Roll speed, feeding rate, and hydraulic pressure were 1.5 rpm, 25 rpm, and 28.44 MPa, respectively. The waste forms were manufactured by incorporating created pellets (H 6.5 × W 9.4 mm) using polymers as solidification agents. Used polymers were main ingredient (YD-128), hardener (G-1034), and diluent (LGE). The optimum mixing ratio was YD-128 : G-1034 = 65 : 35 phr, and LGE was added in an amount of 10wt% of the total mixture. To confirm the disposal suitability of the manufactured waste forms, characterization evaluation was carried out (compressive strength, thermal cycling, immersion, and leaching test). Characterization evaluation revealed a minimum compressive strength of 23.1 MPa, far exceeding 3.44 MPa of the disposal facility waste acceptance criteria. Compressive strength increased to the highest value of 31.90 MPa after immersion test. To examine leaching characteristics, the pH, Electrical Conductivity (EC) and leachability index (L_i) of leachates were identified. As results, pH and EC consistently increased or remained constant with leaching time. The average of Co, Cs and Sr nuclides was 17.76, 17.38 and 14.04, respectively, exceeding the value of 6 in the waste acceptance criteria. Effective waste treatment/disposal can be achieved without increasing volumes of sludge/slurry by enhancing the technique of this research by performing additional studies in the future.

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Keywords: Soil washing, Contaminated fine soil, Roll compaction, Pelletization, Polymer waste form, Characterization

Vitrification Analysis of Washing Sludges From Stainless Steel Tank Using Hot Isostatic Press & High-Temperature Furnace

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The optimum vitrification conditions of the radioactive waste using high-temperature furnace and HIP (Hot Isostatic Press) were studied for the successful reduction of the solidification volume, radioactive level, satisfying the disposal criteria such as leaching rate and compressive strength. Vitrification is receiving attention for the solidification disposal of intermediate and low-level radioactive wastes for its chemical-physical stability and leachability. Its principle is to trap the radioactive material in a fixed structure of the glass type materials, such as Boron Trioxide, Silicon Dioxide and Phosphorus Pentoxide. Sludge targeted in this study is assembly of materials while sludge is stored in the stainless-steel tank before disposal, which consists of Fe_3O_4 (14.9wt%), Fe_2O_3 (3.8wt%), and Cr_2O_3 (6.3wt%), cement paste (25wt%) and detergent/shower sludge (50wt%). The detergent/shower sludge generated from the washing the clothes that were worn during the work at the laboratory and nuclear power plant contains organic materials that are vulnerable to chemical reactions, therefore, immobilization of organic material by the incinerating step, which can also immobilize the radioactive substance, was applied. Its composition – containing Cs-133 and Co-59 substitution for Cs-134 and Co-60 that are radioactive – was analyzed by XRD before and after the mineralization of the sludge using high temperature furnace in different temperature, to identify the remaining element and the features of the mineralized sludge. Targeted sludge was vitrified using Hot Isostatic Press in with different pressure and temperature conditions, to find out the optimum vitrification conditions. Vitrified waste was evaluated in many aspects - leaching evaluation following ANS16.1, compressive strength evaluation of 3.44 MPa (waste disposal criteria), volume reduction before and after the sequence.

Keywords: Hot isostatic press (HIP), Vitrification, Washing sludge, Immobilization, Mineralization

Effects of Rock Damage Models on the Hydro-Mechanical-Damage Coupled Analysis of Radioactive Waste Disposal Structures

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Low to intermediate radioactive waste disposal concrete structures are subjected to coupled hydro-mechanical conditions and the identification of structural damage is crucial to ensure safe long-term disposal. Different damage models for concrete and the surrounding rock can affect the damage characteristics of radioactive waste disposal structures. In this study, the effects of different rock damage models are applied to the hydro-mechanical-damage coupled structural analysis of the Wolseong Low and Intermediate Level Radioactive Waste Disposal Center silo. A two-dimensional model of the disposal silo was modeled using the finite element analysis software COMSOL and the Mazars' damage model was applied to the silo concrete. The Mazars' model parameters were obtained from uniaxial compression and tensile tests on cylindrical concrete specimens after 28 days of water curing and further 32 days of wet curing at 75°C). The COMSOL embedded Richards equation module was used to simulate hydraulic analysis. Structural loading due to waste disposal was applied at the bottom of the silo structure and the damage evolution characteristics were investigated. The non-linear mechanical rock behavior obtained from laboratory tests (Hoek-Brown criterion, resonant column test, Mazar's damage model) and field tests (Goodman Jack) were input to assess the effects of different rock damage models. The results highlight the importance of structural damage consideration when assessing the long-term stability and safety of underground radioactive waste disposal structures under coupled hydro-mechanical conditions.

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Keywords: THMD coupled analysis, Rock damage, Radioactive waste disposal, COMSOL

Preliminary Safety Assessment of Disposal of Glass Fiber Waste

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In a recent preliminary inspection for disposal, the glass fiber waste (GFW), used as a pipe insulation, was judged as “pending evaluation” because some dust was found in drum opening tests. Therefore, additional inspection is required to ensure that the package corresponds with the acceptance criteria of the particulates. The dust was generated presumably due to GFW being used in a high-temperature environment for a long time, thus being easily degraded and crushed. For this reason, safety issues that may occur in the process of handling, transportation, and disposal are emerging. Therefore, in this study, a preliminary safety assessment of GFW disposal was performed, the exposure dose to the general public was derived, and compared with the dose limit. The evaluation was carried out in the following order: (1) evaluation of GFW radiation source term, (2) selection of accident scenario, (3) calculation of exposure dose, (4) comparison of evaluation results with dose limits, and confirmation of satisfaction. The average radioactivity of the GFW to be disposed of was used as the source term, and the main nuclides were identified as H-3, Fe-55, Co-60, Ni-63, and Pu-241. In general, the types of accidents that can occur at disposal facilities can be classified into falls, fires, collisions during transportation, off-site accidents, and nuclear criticality, and the accident scenarios are selected by analyzing and reviewing the probability of each accident. In this study, the accident analysis and scenarios presented in the safety assessment of the KORAD were reviewed, and the fire in the treatment facility, the fire in the storage facility, and the collision of the transport vehicle were selected as the evaluation scenarios. When an accident occurs, the radioactive material inside the container leaks out and diffuses into the atmosphere. In this evaluation, the internal and external exposure of the general public due to radioactive plume at the site boundary was evaluated and the dose conversion factors from ICRP-72 and FGR 12 were used. Based on the evaluation, general public was exposed to 0.004 mSv, 0.013 mSv, and 0.045 mSv, respectively, due to a fire at a treatment facility, at a storage facility, and in a transport vehicle. Most of the dose is due to internal exposure by Pu-241 nuclide, because the proportion of it in the waste is high, and when inhaled, the internal dose is high by emitting beta rays. It was confirmed that the result of dose was 0.4%, 1.3% and 4.5% of the annual dose limit, sufficiently satisfying the dose limit and safety.

Keywords: Radioactive waste, Safety assessment of disposal, Glass fiber waste, LILW disposal facility, Dose limit, Waste acceptance criteria

Safety Analysis and Evaluation of Very-Low-Level Liquid Radioactive Waste Transport and Storage Container

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The mechanical safety of the container designed according to the IP-2 type technology standard was analyzed for the temporary storage and transportation of Very-Low-Level-Waste (VLLW) for liquid occurring at the nuclear facilities decommissioning site. The container was designed and manufactured as a composite shielding container with the effect of storing and shielding liquid radioactive waste using High Density Polyethylene (HDPE) and eco-friendly shielding material (BaSO₄) with corrosion and chemical resistance. The main material of the composite shielding container is HDPE and BaSO₄, the material of the cover, cage and pallet is SUS304, and the angle guard is elastic rubber. The test and analysis requirements were analyzed for structural analysis of container drop and lamination test. As test requirements for IP-2 type transport containers should be verified by performing drop and lamination tests. There should be no loss or dispersion of contents through the 1.2 m high free-fall drop and lamination test for a load five times the amount of transported material. ABAQUS/Explicit, a commercial finite element analysis program, was used for structural analysis of the drop and lamination test of the transport and storage container. (Drop test) It was confirmed that the container was most affected when it falls from a 45-degree slope. Although plastic deformation was observed at the edge axis of the cover, it was evaluated that the range of plastic deformation was limited to the cover and cage, and stress within the elastic limit occurred in the inner container. In the analysis results for other falling direction conditions, it was evaluated that stress within the elastic limit was generated in the inner container except for minor plastic deformation. In the case of on-site simulation evaluation, deformation of the inner container and frame due to the drop impact occurred, but leakage and loss of contents, which are major evaluation indicators, did not occur. (Lamination test) The maximum stress was calculated to be 19.9 MPa under the lamination condition for a load 5 times the container weight, and the maximum stress point appeared at the corner axis of the pallet. The calculated value for the maximum stress is about 10%, assuming the conservative yield strength of SUS304 is 200 MPa. It was evaluated that stress within the limit occurred. In the case of on-site simulation evaluation, it was confirmed that there was no container deformation or loss of contents due to the load.

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Keywords: Very-Low-Level liquid radioactive waste, Transport and storage container, Safety analysis, D&D

The Recycling Scenarios of Metal Clearance Wastes Generated From the Decommissioning of Nuclear Power Plants

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In this study, the current situation of recycling domestic and foreign metal clearance waste was reviewed to suggest the optimal recycling scenario for metal clearance waste that occurs the most when decommission nuclear power plants. Factors that can directly or indirectly affect the recycling of metal clearance waste were analyzed and evaluation criteria that can be used to evaluate optimal recycling measures were prepared. Using this, a scenario for recycling the optimal metal clearance waste suitable for the domestic environment was proposed.

As a result of comparing/reviewing the importance of the first level of the evaluation criteria, public acceptance, national policy, and regulatory requirements were evaluated as the most important ones, and recycling acceptance and regulatory requirements were evaluated as the most important the second level of evaluation criteria. As a result of reviewing the clearance waste recycling scenario, it was evaluated that unrestricted recycling scenario was preferred. This may be because the survey subjects are composed of experts in the nuclear power field, so they know recycling of clearance waste in general industries does not significantly affect radiation safety. However even if it is clearance waste, the public may feel reluctant to recycle just because it was discharged from nuclear power plants, so policy and institutional improvements are needed to reassure the public along with the scientific safety of clearance waste. In addition, in order to improve public acceptance, it seems necessary to prepare specific measures to ensure the participation of public in the entire decommissioning process, share related information, and disclose all routes from generation to disposal of decommissioning waste.

Considering that research on domestic clearance waste recycling options has not been activated, this study is significant in that it derives a scenario for recycling metal clearance waste that can be implemented. Also, it is expected that the evaluation criteria derived from this study will be used significantly when establishing a radioactive waste management strategy.

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Keywords: Decommissioning, Recycling scenario, Metal clearance waste

Evaluating the Influence and Correlation of EDTA and ISA From the Perspective of the Safety of Disposal Facility

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Safety for the radioactive waste disposed of in the disposal facility should be secured through safety assessment in consideration of the various situations. In this study, the influence and correlation of EDTA and ISA, which are the factors that can impede the safety of the disposal facility, were analyzed using the PHREEQC computational code. Thermodynamic database (TDB) of Andra, specific ion interaction theory (SIT) model as ionic strength correction model, radionuclides (Ni, Am, Pu) were adopted to perform the calculation on the distribution of chemical species by pH. According to the results, EDTA dominated the system and the effect of ISA is relatively small for the distribution of the chemical species of divalent and trivalent cations in neutral and weak base conditions. In the case of the tetravalent cations, the effect of ISA increased compared to the previous case especially in the strong base conditions. In conclusion, EDTA has a more significant effect on the system than ISA under the environment of the domestic disposal facility. Furthermore, when EDTA and ISA are present simultaneously in the system, the effects of two materials are inversely proportional and this characteristic should be considered during the safety assessment.

Keywords: Safety assessment, Radioactive waste, EDTA, ISA, PHREEQC

Modeling Technique for Concrete Target in Drop Analysis of Disposal Container for Radioactive Wastes

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In this study, a drop analysis of metallic disposal containers for radioactive wastes is performed according to accident scenarios at the disposal site. The weight of the disposal container is about 8 tons, and the ingot-type wastes are loaded in the disposal container. To simulate the floor of the disposal site as the impact target, the reinforced concrete pad is modeled. High impact energy of the disposal container due to their heavy weight and high drop height causes excessive deformation and failure of the concrete target having relatively weak strength. Dynamic growth of cracks due to such failures causes penetration and delamination of concrete. Since the impact force delivered to the container strongly depends on the failure of the concrete pad, it is important to properly simulate the failure of the concrete in the drop analysis. A material erosion method can be used to simulate the concrete failure. In the case of applying erosion based on the finite element method (FEM), the element is deleted when the element exceeds a certain criterion, which causes material and energy loss problem. To solve this problem, mesh-free methods such as smoothed particle hydrodynamics (SPH) can be commonly used, but the mesh-free method has the disadvantage of incurring high numerical cost. Therefore, an adaptive method combining SPH and FEM-based SOLID elements is used for concrete target modeling to simulate excessive deformation and failure of the concrete target. In the adaptive coupling method of SPH and SOLID, the concrete target is first modeled as a solid element. When the damage of concrete exceeds the failure criterion, the solid element is eroded and the SPH element replacing the solid element is activated. Since the activated SPH element continues to participate in the impact, the problem of loss of materials and energy can be effectively solved. In this way, analysis results consistent with actual physical phenomena can be obtained.

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Keywords: Disposal container, Radioactive waste, Drop test, Accident scenario

A GoldSim Model for the Radionuclide Migration of LILW Disposal Site in Gyeongju

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The chelating agent and cellulose generated during the operating and decommissioning of a NPP's form organic complexing compounds. That is accelerate the migration of radionuclide and have a bad influence of LILW disposal site. In this study, the GoldSim (RT module) program was used for the effects of radionuclide migration by organic complex compounds as described above. A scenario was derived for evaluation, and a conceptual design (Concept Art) of the GoldSim model was performed. 1) Derivation of the scenario. For the scenario, we selected a groundwater flow scenario in which groundwater flows in and radionuclides flow out after a lapse of time after the operation of the LILW disposal site in Gyeongju is closed. The inflowing groundwater comes into contact with radioactive waste and the radionuclides dissolve. The dissolved nuclides move past the drum and out of the disposal vessel due to the advection phenomenon. Radionuclides spilled from the disposal vessel pass through the silo internal filler (crushed stone) and reach the engineering barrier concrete. Radionuclides from degraded concrete are scenarios that move along the flow of groundwater to the near and far. 2) Radionuclide migration concept design. The radionuclide movement section was largely designed with Inner (Inside the silo), Near and Far. (A) Inner (Inside the silo) This section is where radionuclides move from the radiation source to the engineering barrier (silo). The detailed migration path was designed to allow radioactive nuclides to flow out and move to waste drums, solidified matrix of in-drum, disposal vessel fillers, disposal vessels, silo fillers (crushed stones), and engineered barriers (concrete). The LILW disposal site in Gyeongju has a total of 6 silos. Each of the 6 silos was modeled and designed in consideration of the structural information and positional impact. (B) Near & Far. In generally design, the near is form source term to engineered barrier and far is beyond the engineered barrier. In this study, the near and far designed by radionuclide in the section from the beyond the engineering barrier (silo) to the sea through the groundwater flow through the natural rock. Especially in the case of near, the design was made by applying the position of the natural rock sampling drill hole.

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Keywords: Organic complexing agents, Organic complexing compounds, Treatment technology, Radioactivewaste acceptance criteria, Radionuclides migration, Disposal safety

Application of DIC Method to Drop Test of Transport Container for Radioactive Wastes

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For transport containers for radioactive wastes, a drop test should be performed at a height of 0.3–1.2 m on a rigid target depending on the weight as a normal condition in the regulation. In the drop test, a strain gauge is commonly used to measure the local strain, and the position of the strain gauges is determined by the experiences of the engineer in advance of the test. For this reason, the strains can be measured at only predetermined points. The DIC (Digital Image Correlation) method using high-speed cameras can be used to measure the change in strain over the region of interest. In addition, it is possible to measure effectively even in areas with high strain gradients that are difficult to measure with strain gauges. Therefore, the DIC method can measure the strain change according to time over the entire load path. When the drop test of the transport container is performed, the impact load is delivered through the lower corner fittings-corner posts-upper corner fittings-lids. In this study, white spray was sprayed on these main load path, and black speckles were created on the sprayed surface to trace the rigid motion of speckles. The images taken during the drop test can be used to create a strain field over region of interest.

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Keywords: Transport container, Drop test, Digital image correlation method

Effect of Specimen Size on Acoustic Signals for Repository Monitoring

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Low and intermediate radioactive wastes in South Korea have been disposed in Wolsong Low and Intermediate Level Radioactive Waste Disposal Center (WLDC), Gyeongju. This repository structure is planned to be operated few hundred years while toxicity of the waste is sufficiently decayed. The structural integrity of the repository is required to protect the waste in safe. The integrity of the structure is commonly estimated using acoustic emission (AE) method. The integrity of the structure using AE is obtained by following process: 1) Estimation of maximum acoustic crack energy of the structure, 2) Acoustic signal measurement and filtering, and 3) Measurement of simultaneous acoustic cracking energy. The damage of the structure can be obtained from cumulative cracking energy from the structure divided by the predicted maximum cracking energy of the structure. Estimation of maximum cracking energy is gained by the specimens whose components are identical to the repository structure. The cracking energy of the different specimens are obtained during uniaxial compressive test and volume of the specimen is calculated. Then, the fractal coefficient for the structure is obtained and the maximum crack energy of the target structure can be calculated. The specimens whose diameters vary from 50 mm to 150 mm and heights are twice of the diameter are made with same recipe of WLDC silo concrete. The uniaxial compression test is conducted with loading rate of $0.1 \text{ mm} \cdot \text{min}^{-1}$. The fractal coefficient is obtained by least square method from the volume-cumulative energy relationship.

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Keywords: Low and intermediate radioactive waste, Disposal repository, Fractal theory, Scale effect, Acoustic emission signal, Uniaxial compression test

Effective Treatment of Uranium-contaminated Soil-washing Effluent Using Precipitation/distillation Process for Water Reuse and Self-disposal

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The purpose of this study was to effectively purify U-contaminated soil-washing effluent using a precipitation/distillation process, reuse the purified water, and self-dispose of the generated solid. The U ions in the effluent were easily removed as sediments by neutralization, and the metal sediments and suspended soils were flocculated–precipitated by polyacrylamide (PAM). The precipitate generated through the flocculation–precipitation process was completely separated into solid–liquid phases by membrane filtration (pore size < 45 μm), and Ca^{2+} and Mg^{2+} ions remaining in the effluent were removed by distillation. Even if neutralized or distilled effluent was reused for soil washing, soil decontamination performance was maintained. PAM, an organic component of the filter cake, was successfully removed by thermal decomposition without loss of metal deposits including U. The uranium concentration of the residual solids after distillation is confirmed to be less than $1 \text{ Bq}\cdot\text{g}^{-1}$, so it is expected that the self-disposal of the residual solids is possible. Therefore, the treatment method of U-contaminated soil-washing effluent using the precipitation/distillation process presented in this study can be used to effectively treat the washing waste of U-contaminated soil and self-dispose of the generated solids.

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Keywords: Uranium, Soil washing, Precipitation, Distillation, Self-disposal

Global Sensitivity Analysis of the Safety Assessment Model for Wolsong LILW Disposal Facility

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To obtain confidence in the safety of disposal facilities for radioactive waste, it is essential to quantitatively evaluate the performance of the waste disposal facilities by using safety assessment models. Thus, safety assessment models require uncertainty management as a key part of the confidence-building process. In application to the numerical modelling, the global sensitivity analysis is widely employed for dealing with parametric and conceptual uncertainties. In particular, the parametric uncertainty can be effectively reduced by minimizing the uncertainty of critical parameters in the safety assessment model. In this paper, the numerical model of each step disposal facility (Silo, Near surface, and Trench type) at Wolsong Low and Immediate Level Waste (LILW) Disposal Center is designed by using a two-dimensional finite element code (COMSOL Multiphysics). In order to determine the critical parameters for non-adsorbed nuclides such as H-3, C-14, Tc-99, we introduced the variance-based sensitivity analysis methodology of the global sensitivity analysis. In the case of Silo type, the density of waste is highly sensitive to the total leakage quantity of all nuclides. Additionally, the initial nuclide concentration of H-3 was identified as another important parameter of H-3. On the other hands, the mass transport coefficient showed a high contribution in C-14 and Tc-99. In other types of disposal facilities, the leaking properties of H-3 are significantly affected by the amount of infiltration water. However, C-14 and Tc-99 were found to be more sensitive to the density of waste.

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Keywords: Global sensitivity analysis, Variance-based sensitivity, Disposal facility, Safety assessment model, Radioactive waste

The Thermal Stability Evaluation of Silver Tellurite Glass for Radioactive Iodine Immobilization

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Uranium-235, used for nuclear power generation, has brought radioactive waste. It could be released into the environment during reprocessing or recycling of the spent nuclear fuel. Among the radioactive waste nuclides, I-129 occurs problems due to its long half-life (1.57×10^7 y) with high mobility in the environment. Therefore, it should be captured and immobilized into a geological disposal system through a stable waste form. One of the methods to capture iodine in the off-gas treatment process is to use silver loaded zeolite filter. It converts radioactive iodine into AgI, one of the most stable iodine forms in the solid state. However, it is difficult to directly dispose of AgI itself in an underground repository because of its aqueous dissolution under reducing condition with Fe^{2+} . It must be immobilized in the matrix materials to prevent release of iodine as a result of chemical reaction. Among the matrix glasses, silver tellurite glass has been proposed. In this study, additives including Al, Bi, Pb, V, Mo, and W were added into the silver tellurite glass. The thermal properties of each matrix for radioactive iodine immobilization were evaluated. The glasses were prepared by the melt-quenching method at 800°C for 1 h. Differential scanning calorimetry (DSC) was performed to evaluate the thermal properties of the glass samples. From the study, the glass transition temperature (T_g) was increased by adding additives such as V_2O_5 , MoO_3 , or WO_3 in the silver tellurite glass. The relative electro-static field (REF) values of V_2O_5 , MoO_3 , and WO_3 are about three times higher than that of the glass network former, TeO_2 . It could provide sufficient electro-static field (EF) to the TeO_2 interacting with the non-bridging oxygen forming Te-O-M ($\text{M} = \text{V}, \text{Mo}, \text{W}$) links. Therefore, the addition of V_2O_5 , MoO_3 , or WO_3 reinforced the glass network cohesion to increase the T_g of the glass. The addition of MoO_3 or WO_3 in the silver tellurite glass increased T_g and crystallization temperature (T_c) with remaining the glass stability.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP) (NRF-2021M2E3A1040061).

Keywords: Radioactive waste, Iodine immobilization, Silver tellurite glass, Glass transition temperature, Crystallization temperature, Glass stability

Aerosol Processing of CaCO_3 for C-14 Recycle From PHWR Waste

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In this study, an aerosol process was introduced to produce CaCO_3 . The possibility of producing CaCO_3 by the aerosol process was evaluated. The characteristics of CaCO_3 prepared by the aerosol process were also evaluated. In the CaCO_3 prepared in this study, as the heat treatment proceeded, the calcite phase disappeared. The portlandite phase and the lime phase were formed by the heat treatment. Even if the CO_2 component is removed from the calcite phase, there is a possibility that the converted CO_2 component could be adsorbed into the Ca component to form a calcite phase again. Therefore, in order to remove the calcite phase, carbon components should be removed first. The lime phase was formed when CO_2 was removed from the calcite phase, while the portlandite phase was formed by the introducing of H_2O to the lime phase. Therefore, the order in which each phase formed could be in the order of calcite, lime, and portlandite. The reason for the simultaneous presence of the portlandite phase and the lime phase is that the hydroxyl group (OH^-) introduced by H_2O was not removed completely due to low temperature and/or insufficient heating time. When the sufficient temperature (900°C) and heating time (60 min) were applied, the hydroxyl group (OH^-) was removed to transform into lime phase. Since the precursor contained the hydrogen component, it could be possible that the moisture (H_2O) and/or the hydroxyl group (OH^-) were introduced during the heat treatment process.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korean government (MOTIE) (20201520300140, Development of Advanced Functional Material with C-14 from PHWR Waste).

Keywords: Aerosol process, C-14, Calcium Carbonate, Hydroxyl group

Review on the Confidence Limits in the Mean Activity Method for Indirect Evaluation of Difficult-to-Measure (DTM) Nuclide Concentration in Radioactive Waste

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In order to indirectly evaluate the inventory of difficult-to-measure (DTM) nuclides in radioactive waste, the scaling factor method by key nuclide has been used. It has been usually applied to low-and intermediate-level dry active waste (DAW), and the tolerance of 1,000% margin of error in the US, that is the factor of 10, is applied as an allowable confidence limits considering the inhomogeneity of the waste and the limitation of sample size. This is because the scaling factor method is based on economic efficiency. Confidence limits is the uncertainty (sampling error) according to predicting the mean value of the population by the mean value of the sample at 95% confidence level, reflecting the limitations of sample size (representation) with the standard deviation. If the standard deviation is large, the sample size can be increased to satisfy the allowable confidence limits. In the new nuclear power plants, the concentration of cesium nuclide (^{137}Cs) in radioactive waste tends to be very low due to advances in nuclear fuel and reactor core management technology, which makes it very difficult to apply cesium as a key nuclide. In addition, it is inevitable to apply the mean activity concentration method, which reasonably and empirically derives the concentration of DTM nuclides regardless of key nuclide, when the correlation between key and DTM nuclides is not significant. The mean activity method is a methodology that applies the average concentration of a sample set to the entire population, and is similar to applying the average concentration ratio between key and DTM nuclides of a sample set to the population in the scaling factor method. Therefore, in this paper, the maximum acceptable uncertainty (confidence limits) at a reasonable level was studied when applying the mean activity concentration method by arithmetic mean unlike the scaling factor method which usually uses the geometric mean method. Several measures were proposed by applying mutatis mutandis the acceptable standard deviation in radiation measurement and the factor of 10 principle, etc., and the appropriateness was reviewed through case analysis.

Keywords: Mean activity method, Confidence limits, Uncertainty, Difficult-to-measure (DTM) nuclide, Indirect evaluation

The Improvement of the Off-site Dose Calculation Program to Enhance the Efficiency and Reliability of the Evaluation Process

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The off-site dose calculation is regularly carried out at the nuclear power plants in order to evaluate off-site dose from gaseous and liquid effluent during normal operation. In 2009, the off-site calculation program (K-DOSE60) was developed in accordance with ICRP-60 by KHNP. This software needs meteorological data, gaseous and liquid effluent data, and various other input parameters to evaluate off-site dose. As a result, it takes a certain amount of time for the user to enter accurate input data and verify calculated results, and it is difficult to intuitively determine them because of providing text-based calculated results. Therefore, in this study, the improvement of the calculation program was considered so that a more reliable and effective evaluation could be performed when calculating the off-site dose.

The main improvements of the off-site dose calculation program (ODCP) are as follows. First, it is developed as the network-based program to link with meteorological data, and gaseous and liquid effluent data to remove input errors and simplify data transfer. Second, through validation process of input data, input errors are eliminated. Third, the input data and calculated results are visually provided so that the user can easily determine the evaluation results. Fourth, database of input and calculated results is constructed to facilitate evaluation result history management.

Keywords: Off-site dose calculation program, Network-based program, Nuclear power plant

Self-disposal of MCAs in Radwaste Form Test Facility for Reuse

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The radwaste facility management team is preparing for clearance of 4 MCAs in The Radwaste Form Test Facility (RFTF). The targeted waste was used for clearance level radioactive waste sample analysis and has been used for this purpose since the early 2000s. Due to the characteristics of clearance level radioactive waste, the concentration of radioactivity is very low and MCA is used with Marinelli beakers the possibility of contamination is low. Moreover, the radiation detector should not be contaminated with radioactive materials, it should be less than the clearance level.

These detectors were considered surface contamination materials. To detect the contaminated spot of each material, we scanned the whole surface of a material with a gamma survey meter. After that, we took a sample from $1 \times 1 \text{ m}^2$ and a total of 30 samples from each MCA. The wiped filter paper was analyzed with alpha, beta low background counting systems. The results of the analysis of the smear sample of total alpha and beta nuclide radioactivity were less than MDA (α : $2.88 \times 10^{-5} \text{ Bq} \cdot \text{cm}^{-2}$, β : $3.07 \times 10^{-5} \text{ Bq} \cdot \text{cm}^{-2}$).

The major nuclide in this facility is Co-60 and Cs-137 therefore we analyzed gamma nuclide activity with HPGe. The maximum specific activity was Co-60: $2.31 \times 10^{-5} \text{ Bq} \cdot \text{cm}^{-2}$, Cs-137: $1.96 \times 10^{-6} \text{ Bq} \cdot \text{cm}^{-2}$. If it is satisfied with the clearance criteria, detectors will be reused at the Radioactive Waste Treatment Facility (RWTF) room # 7251 uncontrolled area.

Keywords: Self-disposal, Clearance, Reuse, Radioactive waste, MCAs

CNN-based Approach for Source Localization of an Engineering-scale Radioactive Waste Disposal Concrete

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Source localization technique using acoustic emission (AE) has been widely used to track the accurate location of the damaged structure. The principle of localization is based on signal velocity and the time difference of arrival (TDOF) obtained from different signals for the specific source. However, signal velocity changes depending on the frequency domain of signals. In addition, the TDOF is dependent on the signal threshold which affects the prediction accuracy. In this study, a convolutional neural network (CNN)-based approach is used to overcome the existing problem. The concrete block corresponding to $1.3 \times 1.3 \times 1.3$ m size is prepared according to the mixing ratio of Wolseong low-to-intermediate level radioactive waste disposal concrete materials. The source is excited using an impact hammer, and signals were acquired through eight AE sensors attached to the concrete block and a multi-channel AE measurement system. The different signals for a specific source are time-synchronized to obtain TDOF information and are transformed into a time-frequency domain using continuous wavelet transform (CWT) for consideration of various frequencies. The developed CNN model is compared with the conventional TDOF-based method using the testing dataset. The result suggests that the CNN-based method can contribute to the improvement of localization performance.

Acknowledgements

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Keywords: Source localization, Radioactive waste disposal concrete, Convolutional neural network, Acoustic emission

~~Study on B-Type Cask for Transportation and Storage of Radioactive Materials~~

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Canceled Poster

Design of Mobile Plasma Melting System for Non-combustible Wastes

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In this work, we introduce a 100 kW class mobile plasma melting system designed for non-combustible radioactive wastes treatment. To ensure mobility, the designed system consists of two 24-ft commercial containers, each in charge of the plasma utilities and melting process. In the container for plasma utilities, a 100 kW class DC power supply is installed together with a chiller and gas supply system whereas the container for melting process has a transferred type arc melter as well as off-gas treatment system consisting of a heat exchanger, filtrations, scrubber and NOx removal system. As a heat source for a transferred type arc melter, we adopted a hollow electrode plasma torch with reverse polarity discharge structure. Detailed design for a 100 kW class mobile plasma melting system will be presented together with the main specifications of the components. In addition, the basic performance data of the melting system is also presented and discussed.

Keywords: Plasma melting, Non-combustible, Radioactive waste, Mobile system

Extract Calcium Hydroxide From Dismantled Waste From Nuclear Power Plants and Recycle it

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As the number of aging nuclear power plants increases, the market for dismantling nuclear power plants is growing rapidly. About 40% of the cost of dismantling nuclear power plants is the waste treatment cost incurred during the dismantling process, of which concrete waste accounts for a significant portion of the total waste. Securing technology for reducing and recycling concrete waste is very important not only in terms of economy but also in terms of environment.

The objective is to synthesize geopolymer using inorganic materials from cement fine powder in concrete waste. Cement fine powder in concrete waste has a large amount of inorganic elements necessary for filling materials for radioactive waste treatment such as CaO and SiO₂. In particular, Ca(OH)₂ is synthesized by extracting Ca²⁺ from concrete waste. It can be used as an alkali activator to synthesize geopolymer.

The mortar from crushed concrete was used as a source of calcium. The first step is to react with concrete waste and hydrochloric acid to extract ions. The second step is to react with NaOH and synthesize Ca(OH)₂. The product was divided into two stages according to the reaction method and order. The first and second products were washed and dried, and then XRD and XRF were performed. The second product was matched only Ca(OH)₂ and CaCO₃ at the XRD peak. In the case of XRF, it was analyzed to have a purity of 67.80–78.73%.

Synthesis of geopolymer by recycling materials extracted from concrete waste can reduce disposal costs and improve the utilization rate of disposal sites.

Acknowledgements

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

Study on Measures to Evaluate the Inventory of Radionuclides in Spent Filter Packaging Drums

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Currently, in domestic nuclear power plants (NPP), the spent filters (SFs) used for the purpose of reducing and purifying the radiation of the primary cooling water system are temporarily stored in an untreated state. In order to dispose of SFs, radioactive nuclide analysis (RNA) of SFs is required to be conducted. As segmented gamma scanner (SGS) is already being used in Kori NPP, utilizing SGS for RNA of SFs would be practical and economical. In this paper, factors required to be considered to improve accuracy of SGSs for RNA of SFs are studied.

The analysis of the nuclide inventory of the packaging drum for radioactive waste should be performed by the indirect drum nuclide analysis method. The material of the SFs is iron (SS304) on the outside, and paper on the inside. In addition, to meet disposal acceptance criteria, radioactive waste drums are packaged in thick grouting or shielding drums. Therefore, it is necessary to derive an appropriate correction method for high inhomogeneity and thick media.

Considering these factors, evaluating radionuclides inventory plans to measure gamma rays in SGS mode. Correct the gamma ray measurement by examining the medium attenuation factor and error factors. In this way, the inventory of gamma nuclides is calculated, and the specific radioactivity of beta ray and alpha particle emitting nuclides other than gamma rays is planned to be calculated by applying scaling factors.

Keywords: Radioactive spent filter, Radionuclides inventory, SGS

5분과

제염해체(Oral)

Lessons Learned From Implementing the Radiological Characterization of Kori 1: Lab Analysis and In-situ Measurement

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An application of the final decommissioning plan for unit 1 of Kori NPP was submitted to NSSC on 14 May 2021. We have been implementing the project related to the radiological characterization for the plan since 2019. However, the project was not running smoothly due to the regulatory environment. The destructive sampling from the objects was not allowed, so only smear (swipe) samples are available. In this study, the sampling way and the analytical results of radionuclides are presented. In addition, we propose in-situ measurement using gamma camera and in-situ gamma spectroscopy to obtain more comprehensive radiological information on the object.

Keywords: Radiological characterization, Kori, Radioactivity, In-situ measurement, Decommissioning

Design of a Pure Paraffin Separating Machine From Concentrated Waste Solidified With Paraffin by Using Heating Vaporization and Pressure Difference

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In domestic nuclear power plants, drums of concentrated radioactive waste solidified with paraffin that do not meet radioactive waste disposal standards are stored temporarily. In this paper, the design of a machine that separates these paraffin drums into paraffin and concentrated waste using heating vaporization and pressure difference is described. The separation process is as follows. First, the paraffin solid is indirectly heated by heating the outside of the drum. The paraffin solid is partially melted to increase the fluidity and is easily detached from the drum. The detached solid is transferred to the melting tank, and further heated in the melting tank. When the temperature is sufficiently high, paraffin is melted and becomes a mixture of liquid paraffin and concentrated waste homogeneously. The mixed solution is transferred to a paraffin recovery vessel and further heated. The vaporization point of paraffin is 370°C under atmospheric pressure, and becomes lower depending on the pressure decreasing in the vessel. The vaporization point of the paraffin is a relatively low value compared to the radioactive elements in the concentrated waste, and therefore only paraffin would be vaporized. A paraffin transfer pipe is installed on the upper part of the paraffin recovery vessel, and is connected to another tank called the paraffin capture vessel. The pressure of the paraffin capture vessel is reduced (i.e. vacuum condition), only gaseous paraffin is transferred to the paraffin capture vessel by the pressure difference. When the paraffin capture vessel is cooled below the vaporization point of the paraffin, the paraffin is liquefied or solidified, and only the paraffin is recovered.

Based on the above process, the solidified paraffin could be separated into pure paraffin and concentrated waste. However, if a radioactive element with a lower vaporization point than paraffin exists in the concentrated waste, it may be mixed with paraffin and separated together. Therefore, it is necessary to measure the radioactivity or radiation dose rate for the separated paraffin, and to verify that it is sufficiently low. If necessary, additional separation process may be considered for removing radioisotopes from the paraffin.

Keywords: Concentrated waste, Solidified paraffin, Separating machine, Machine design, Heating evaporation, Pressure difference

Characterization of Fine and Ultrafine Particles Generated During the Dismantling of a Mock-up Reactor Pressure Vessel

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Cutting reactor pressure vessels (RPV) into acceptable sizes for waste disposal is a key process in dismantling nuclear power plants. In the case of Kori-1, a remote oxyfuel cutting method has been developed by Doosan Heavy Industry & Construction to dismantle RPVs. Cutting radioactive material, such as RPV, generates a large number of fine and ultrafine particles incorporating radioactive isotopes. To minimize radiological exposure of dismantling workers and workplace surface contamination, understanding the characteristics of radioactive aerosols from the cutting process is crucial. However, there is a paucity of knowledge of the by-products of the cutting process.

To overcome the limitations, a mock-up RPV cutting experiment was designed and established to investigate the characteristics of fine and ultrafine particles from the remote cutting process of the RPV at the Nuclear Decommissioning Center of Doosan Heavy Industry & Construction. The aerosol measurement system was composed of a cutting system, purification system, sampling system, and measurement device. The cutting system has a shielding tent and oxyfuel cutting torch and remote cutting robot arm. It was designed to prevent fine particle leakage. The shielding tent acts as a cutting chamber and is connected to the purification system. The purification system operates a pressure difference by generating an airflow which delivers aerosols from the cutting system to the purification system. The sampling system was installed at the center of the pipe which connects the shielding tent and purification system and was carefully designed to achieve isokinetic sampling for unbiased sampling. Sampled aerosols were delivered to the measurement device. A high-resolution electrical low-pressure impactor (HR-ELPI+, Dekati) is used to measure the size distribution of inhalable aerosols (Aerodynamic diameter: 6 nm to 10 μ m) and to collect size classified aerosols.

In this work, the mock-up reactor vessel was cut 3 times to measure the number distribution of fine and ultrafine particles and mass distribution of iron, chromium, nickel, and manganese. The number distribution of aerosols showed the bi-modal distribution; two peaks were positioned at 0.01–0.02 μ m and 0.04–0.07 μ m respectively. The mass distribution of metal elements showed bi-modal and tri-modal distribution. Such results could be criteria for filter selection to be used in the filtration system for the cutting process and fundamental data for internal dose assessment for accidents. Future work includes the investigations relationships between the characteristics of the generated aerosols and physicochemical properties of metal elements.

Acknowledgements

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Keywords: Decommissioning, Radioactive metal waste, Radioactive aerosol, Size distribution

A Study on Gas Flow Visualization Inside the Thermal Cutting Kerf

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In thermal cutting process, gas flow injected from the nozzle has a significant effect on the cutting materials. The gas flow is difficult to observe gases are transparent, therefore, in this study, Schlieren method was adopted to visualize the gas flow inside the kerf. The kerf shape was inserted between two slices of transparent glass in order to imitate the real cutting environment. In order to get the flow characteristics, a high speed camera was equipped and the image processing was applied to compare the before and after injection images. As a result, the method for visualizing gas flow was successfully developed and also expected to be applied to the analysis of purging gas in various welding environments.

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Keywords: Thermal cutting, Gas flow, Visualization, Schlieren method, Image processing

Remote Dismantling System Using Process Planning and Offline Programming for Nuclear Facility Decommissioning

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The remote dismantling system proposed in this paper is a system that performs the actual dismantling process using the process and program predefined in the digital manufacturing system. The key to the successful applying this remote dismantling system is how to overcome the problem of the difference between the digital mockup and the actual dismantling site. In the case of nuclear facility decommissioning, compensation between the virtual world and the real world is difficult due to harsh environments such as unsophisticated dismantling sites, radiation, and underwater, while offline programming can be proposed as a solution for other industries due to its sophisticated and controllable environment. In this paper, the problem caused by the difference in the digital mockup is overcome through three steps of acquisition of 3D point cloud in radiation and underwater environment, refraction correction, and 3D registration. The 3D point cloud is acquired with a 3D scanner originally developed in our laboratory to achieve 1 kGy of radiation resistance and water resistance. Refraction correction processes the 3D point cloud acquired underwater so that the processed 3D point cloud represents the actual position of the scanned object. 3D registration creates a transformation matrix that can transform a digital mockup of the virtual world into the actual location of a scanned object at the dismantling site. The proposed remote dismantling system is verified through various cutting experiments. In the experiments, the cutting test object has a shape similar to the reactor upper internals and is made of the same material as the reactor upper internals. The 105 successful experiments demonstrate that the proposed remote dismantling system successfully solved the key problem presented in this paper.

Keywords: Remote dismantling system, Digital manufacturing system, 3D scanner, Refraction correction, 3D registration

Comparison and Analysis of the Quantity ^{60}Co Radioactivity Removed and the Radwaste Generated After Full System Decontamination of Nuclear Power Plants

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The purpose of full system decontamination before decommissioning a nuclear power plant is to reduce radiation exposure of decommissioning workers and to reduce decommissioning waste. In general, full system decontamination removes the CRUD nuclides deposited on the inner surface of the reactor coolant system, chemical and volume control system, residual heat removal system, pressurizer, steam generator tube, etc. by chemical decontamination method. The full system decontamination process applied to Maine Yankee and Connecticut Yankee in the USA, Stade, Obrigheim, Unterweser, Nekawestheim Unit 1 in Germany, Mihama Unit 1 and 2 in Japan, Jose Cabrera Unit 1 in Spain, and Barseback Unit 1 and 2 in Sweden are HP/CORD UV, NP/CORD UV, and DfD. In this study, the quantity of ^{60}Co radioactivity removal, metal removal, ion exchange resin and filter generation according to reactor power, surface area and volume of the full system decontamination flow path, and the decontamination process were compared and analyzed. In addition, the quantity of ^{60}Co radioactivity removal by each nuclear power plant was compared and analyzed with the evaluation results of the ^{60}Co radioactivity inventory of the Kori Unit 1 full system decontamination loops conducted by SAE-AN Enertech Corporation.

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Keywords: Full system decontamination, Decommissioning, CRUD, Radwaste, Radioactivity inventory

Cost Evaluation for Transport of the Decommissioning Waste of Kori 1 According to Transport Modes

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Radioactive waste generated during the decommissioning of Kori Unit 1 can be packaged in a transport container under development and transported to a disposal facility by sea transport or land transport. In this study, the cost of each transport method was evaluated by considering the methods of land transport, sea transport, and parallel transport of the radioactive waste dismantled at Kori Unit 1.

In evaluating the shipping cost, the shipping cost was evaluated by assuming the construction of a new ship without considering shipping by CHEONG JEONG NURI, which is currently carrying operational waste. Since the cargo hold of CHEONG JEONG NURI was built to fit the existing operating waste transport container and is not suitable for transporting the transport container currently under development, sea transport using CHEONG JEONG NURI was excluded in this paper.

In the case of on-road transportation, the final fare for each distance was calculated in accordance with the Enforcement Decree of the Freight Vehicle Transportation Business Act, and the cost of on-road transportation was evaluated by estimating the labor cost of the input manpower required for on-road transportation.

The cost of on-road transportation was estimated to be approximately KRW 510 million, the product of the total number of transports 459 times the sum of the cost of transportation vehicle freight cost of about KRW 720,000 and the labor cost of input personnel of KRW 380,000.

It is difficult to predict the cost of building a new ship at this point, as the cost of building new ship is determined by the cost of number of items such as ship design specifications and material prices, labor costs, and finance costs at the time of construction. Accordingly, considering the 2% annual inflation rate based on the shipbuilding cost (about KRW 26 billion) and financing cost (about KRW 12 billion) at the time of construction of the CHEONG JEONG NURI (2005 yr.), decommissioning of Kori Unit 1 (2025 yr.) construction cost finance cost was estimated and evaluated.

According to the result of comparing the transport cost for each transport scenario, land transport is about 510 million won, which is advantageous in terms of economic feasibility compared to the sea transport scenario. However, when transporting by land, it is disadvantageous in terms of acceptability of residents because it is transported multiple times on general roads.

The cost of building a new ship is about KRW 56.4 billion, which is disadvantageous in terms of the cost of transporting waste from the dismantling of Kori Unit 1. But, in the future, cost reduction can be expected if waste materials issued when dismantling nuclear power plants are transported.

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Keywords: Transport method, Cost evaluation, Transport container, On-road transportation, Marine transportation

Study on the Development of Contaminated Soil Decontamination Process on the Site of Nuclear Facilities Using Supercritical Carbon Dioxide

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Today, the domestic and international nuclear power industry is experiencing an acceleration in the scale of the nuclear facility decommissioning market. This phenomenon is also due to policy changes in some countries, but the main reason is the rapid increase in the proportion of old nuclear power plants in the world, mainly in countries that introduced nuclear power plants in the early stages. Decontamination is essential in the process of decommissioning nuclear facilities. Among various decontamination targets, radionuclides are adsorbed between pores in the soil, making physical decontamination quite difficult. Therefore, various chemical decontamination technologies are used for contaminated soil decontamination, and the current decontamination technologies have a problem of generating a large amount of secondary wastes.

In this study, soil decontamination technology using supercritical carbon dioxide is proposed and aimed to make it into a process. This technology applies cleaning technology using supercritical fluids to decontamination of radioactive waste, it has important technical characteristics that do not fundamentally generate secondary wastes during radioactive waste treatment. Supercritical carbon dioxide is harmless and is a very useful fluid with advantages such as high dissolution, high diffusion coefficient, and low surface tension. However, since carbon dioxide, a non-polar material, shows limitations in removing polar and ionic metal wastes, a chelating ligand was introduced as an additive. In this study, a ligand material that can be dissolved in supercritical carbon dioxide and has high binding ability with polar metal ions was selected. In addition, in order to increase the decontamination efficiency, an experiment was conducted by adding an auxiliary ligand material and ultrasonic waves as additives.

In this study, the possibility of liquefaction of chelating ligands and auxiliary ligands was tested for process continuity and efficiency, and the decontamination efficiency was compared by applying it to the actual soil classified according to the particle size. The decontamination efficiency was derived by measuring the concentration of target nuclides in the soil before and after decontamination through ICP-MS. As a result of the experiment, it was confirmed that the liquefaction of the additive had a positive effect on the decontamination efficiency, and a difference in the decontamination efficiency was confirmed according to the actual particle size of the soil. Through this study, it is expected that economic value can be created in addition to the social value of the technology by ensuring the continuity of the decontamination process using supercritical carbon dioxide.

Acknowledgements

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Keywords: Soil decontamination, Supercritical CO₂, Decommissioning, Chelating ligand, Eco-friendly, Site remediation

5분과

제염해체(Poster)

Development of Concrete Reference Material to Improve the Quality of Radioactivity Measurement for Nuclear Decommissioning Waste

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As Kori-1 permanently shut down in Korea, it is expected that a large amount of radioactive waste will be generated during decommissioning of nuclear power plants. Radioactive concrete waste is contaminated up to depth of 100 mm with radionuclides such as ^{137}Cs and ^{60}Co . The radioactive waste should be accurately classified to reduce the cost of disposing of radioactive waste. Therefore, the specific radioactivity of waste must be precisely evaluated by gamma-ray measurements emitted from the radionuclides. In general, the effectiveness of the radioactivity measurement and process is confirmed using certified reference material (CRM) composed of water or agar. However, the decommissioning waste differs from this CRM in apparent density and chemical elements, so the specific radioactivity is underestimated or overestimated. Therefore, reference material composed of the same apparent density and chemical elements as the sample is required to improve the quality of radioactivity measurement. The purpose of this study is to develop a concrete reference material for the nuclear decommissioning waste. The concrete reference material composed of SiO_2 , CaO , and Al_2O_3 were manufactured in compliance with ISO Guide 35. 10 bottles were randomly selected for homogeneity test, and 2 samples for analysis were taken from each bottle. The specific radioactivity was measured using an HPGe detector with an efficiency of 30%. The results of the homogeneity test of ^{137}Cs and ^{60}Co satisfied the requirements of ISO Guide 35. Coincidence summing and self-absorption effects were corrected using the Monte Carlo efficiency transfer code and Monte Carlo N-Particle transport code. The reference values of ^{137}Cs and ^{60}Co in the concrete reference material were evaluated in the range of 1,000–1,100 $\text{Bq}\cdot\text{kg}^{-1}$ and extended uncertainty was around 7%.

Keywords: Decommissioning, Radioactive waste, Radioactivity measurement, QA/QC, Concrete reference material

Establishing Process to Ensure the Disposal Suitability of Spent Filters

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This study established a process to ensure the disposal suitability of spent filters stored in the untreated state in Kori unit 1 and presented the following procedures and requirements for confirming the disposal suitability for each process. The process for securing spent filter disposal suitability consists of collecting spent filters, compression, immobilization, analysis and packaging, and storage stages. The requirements for confirming the acceptance criteria for each process are as follows. (1) Collecting: Since the high radioactivity spent filters are being stored in the filter room of Kori unit 1, those are collected by a remote system to minimize the exposure dose of workers due to spent filter handling. In order to satisfy the surface dose rate requirements, spent filters with a surface dose rate of $10 \text{ mSv} \cdot \text{hr}^{-1}$ or more are classified and collected, and stored temporary storage place until a separate treatment plan is determined. The checkpoints in this process are the surface dose rate, etc. (2) Compression: The collected spent filters are analyzed gamma nuclides such as Co-60 and Cs-137, using a field-applicable nuclide analyzer, and then applying the scaling factors to determine whether it is disposable. Spent filters whose radioactivity concentration is confirmed to be less than the disposal concentration limit is compressed into compression ratios determined by surface dose rate. The checkpoints in this process are nuclide information, surface dose rate, compression ratio, spent filter loading quantity, etc. (3) Immobilization: A spent filter is a non-homogeneous waste that is immobilized with a proven safety material such as cement if the total radioactivity concentration of nuclides with a half-life of more than 20 years is $74,000 \text{ Bq} \cdot \text{g}^{-1}$. Meanwhile, immobilization of inhomogeneous waste can be considered to satisfy disposal criteria such as particulate matter and filling rate. The checkpoints in this process are the immobilizing material, filling rate, etc. (4) Analysis and Packaging: Immobilized drums shall be determined to be 95% or more of the total radioactivity of waste packages by measuring the radioactivity concentration of nuclides using a nuclide analysis device. Finally, measure the surface dose rate and surface contamination of the package, and attach the package label recording the identification number, date, total radioactivity, surface dose rate, and surface contamination information to the packaging container. (5) Storage: Packaging containers are moved to and stored in a temporary waste storage or storage area before disposal.

Keywords: Spent filters, Waste acceptance criteria, Disposal suitability, Waste treatment process

A Proposal of the Method for Confirming Disposal Suitability by Reviewing the Characteristics of Spent Filters

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As the decommissioning of Kori Unit 1 progresses, securing technology for treatment and disposal of radioactive wastes that have not been disposed of so far, such as spent filters, is recognized as an urgent task. In this study, a method of confirming the disposal suitability of spent filters was presented by reviewing the waste characteristics as presented in the waste acceptance criteria (WAC). The waste characteristics to be satisfied to ensure disposal suitability of waste are largely classified into general requirements, solidification and immobilization requirements, radiological requirements, physical requirements, chemical requirements, and biological requirements. First, the general requirement is to prove that the prohibited waste form has not been introduced into items related to waste form and packaging, and to confirm the suitability of disposal through step-by-step packaging photos, generation information, X-ray inspection, and visual inspection. Second, in the solidification and immobilization requirements, spent filters are non-homogeneous waste, and if the total radioactivity concentration of nuclides with a half-life of more than 20 years is $74,000 \text{ Bq} \cdot \text{g}^{-1}$ or more, they must be immobilized. Third, in order to meet the characteristic criteria for nuclides and radioactivity concentration, sampling and scaling factors development are required and based on this, nuclides must be identified and demonstrated to be below the disposal concentration limits. Surface dose rate and surface contamination should be measured in accordance with standardized procedures and disposal suitability should be confirmed through document tests recording the measured values. Fourth, in order to satisfy the physical requirements of the particulate matter and filling rate characteristics, the spent filter must be immobilized, if necessary, thereby ensuring disposal suitability. Meanwhile, free water in the spent filter should be removed through pre-drying and dehydration, and the disposal suitability should be confirmed by applying a test. Fifth, the criteria for chelating agents should be checked for disposal suitability through operation records and component analysis of spent filters, and documents, that can prove harmful substances are removed in advance and no harmful substances are included in the package, should be provided. Lastly, in biological requirements, if the spent filters contain corrosive or infectious substances, they should be removed in advance and disposal suitability should be confirmed by providing documents that can prove that such substances are not included in the package.

Keywords: Spent filters, Waste acceptance criteria, Disposal suitability, Waste characteristics

Considerations for the Development of Waste Certification Programs Applicable to Spent Filter Disposal

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Currently, treatment and disposal suitability verification methods have not been established for radioactive waste, such as spent filters temporarily stored in each plant, so the WCP (Waste Certification Program) can be applied to verify the suitability of non-conforming waste at the site. In this study, WCP components such as certification organizations, certification methods, certification documents, and quality assurance (QA) plan that should be considered when developing WCP applicable to spent filter disposal were reviewed and presented.

First, a certification organization consists of a certification organization that performs certification work, a certification support organization related to waste generation and treatment, and a quality control organization for waste certification. Especially, the support organization should support the implementation of WCP, so that spent filter processing procedures such as generation information management and immobilization can be properly packaged and transported. Second, in identifying the waste characteristics of the certification method, each characteristic identification procedure and certification method of the acceptance criteria should be described, evidence examining the suitability of general, radiological, physical, chemical, and biological requirements, and processes related to measurement and sampling should be established. In identifying characteristics, satisfaction of waste form, free water requirements, and whether it is subject to immobilization should be checked priorly, and a method of confirming particulate matter and securing filling rate when packaging compressed filters should be included. It is very important to develop a technology for verifying the safety and quality of the immobilized material because immobilization of the filters can be a processing method that satisfies various characteristic criteria. Meanwhile, it is essential to collect samples and develop scaling factors to identify the nuclides of filters and prove that they are below the concentration limits. For chemical and biological requirements, the characteristics are identified through generation information documents, corrective actions are taken and documented in case of nonconformance. Third, certification documents should include immobilization procedure manual, characteristic report, and characteristic test manuals such as free water, particulate matter and filling rate, radiation measurement method manual for packages, profile, and generation documents. Fourth, the QA plan should analyze the QA system of the plants, check the QA inspection details, establish general requirements for QA of spent filter disposal, and specify step-by-step certification work QA activities.

In this study, considerations to ensure the disposal suitability at all stages from generation to disposal of spent filter were presented, and development of a WCP could contribute to preventing nonconformance.

Keywords: Spent filters, Waste acceptance criteria, Waste certification program, Disposal suitability

A Study on the Applicability of Dismantling Reactor Pressure Vessel Using the Mechanical Cutting Method

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The dismantling of the reactor pressure vessel has been carried out at a number of commercial nuclear power plants, including the Zion nuclear power plant in the United States and the Stade nuclear power plant in Germany. The dismantling method for the reactor pressure vessel is either in the air or in the water, depending on the utility. In general, a mechanical cutting method is used when dismantling the reactor pressure vessel in the water. And when dismantling a nuclear reactor pressure vessel in the air, the thermal cutting method is applied. However, there is no case of dismantling commercial nuclear reactor pressure vessel by applying a mechanical method in the air.

In this study, when a nuclear reactor pressure vessel is dismantled by applying a mechanical method in the air, the applicability was evaluated by testing it using a demonstration mockup of Kori Unit 1. For the evaluation, the mockup was made in the actual size of Kori Unit 1. Mechanical cutting devices used the band saw and the circular saw. In the test, the cutting of the reactor pressure vessel was performed remotely by reflecting the working conditions of the decommissioning site. The band saw cutting method was applied to vertical cutting, and the circular saw cutting method was applied to horizontal cutting.

In order to dismantle one cut-off piece, mockup test was performed according to a series of dismantling processes, it consists of preparatory work, vertical cutting process, horizontal cutting process, packaging process and finishing work. The cutting speed of the band saw is $3\text{--}10\text{ mm}\cdot\text{min}^{-1}$, and the cutting speed of the circular saw is $2\text{--}4\text{ mm}\cdot\text{min}^{-1}$.

As a result of the test, when the mechanical cutting method was applied, as is known, the kerf width was smaller than when the thermal cutting method was applied. The cut surface showed a clean state without drag lines generated during thermal cutting. However, the working time was much slower than when the thermal cutting method was applied.

Keywords: Dismantling, Reactor pressure vessel, Mechanical cutting, Mockup

Development of Laser Scabbling Technology for Concrete Decontamination

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Laser scabbling experiments were conducted with the aim of developing concrete decontamination technology. Laser scabbling system contains a 6 kW fiber laser (IPG YLS-6000, $\lambda=1,070$ nm) and optical head, which are connected with process fiber (core dia.: 600 μ m, length: 20 m). Optical head consists of two lenses ($f=160$ mm and 100 mm) to collimate and focus laser beam. The focused laser beam is passed through the small diameter of nozzle (throat dia.: 3 mm) to prevent the laser-produced debris into head. And then, the focused beam is directed toward concrete block as continuously diverging. The diverged laser beam was incident on the high-strength concrete with 300 mm (length) $\times$ 300 mm (height) $\times$ 80 mm (width) to induce explosive spalling on the concrete surface. The optical head was moved by X-Y-Z manipulate coupled with a computerized numerical control while scabbling. We have observed not only active spalling on the concrete surface but energetic scattering of laser-produced debris when scabbling on high-strength concretes. It indicates the need for a device capable of collecting the laser-produced debris. We newly designed and manufactured dust collector coupled with cylindrical tube to prevent scattering of laser-produced debris into ambient environment. The collecting system was evaluated by estimating the collecting efficiency for laser-produced debris while scabbling. The collecting efficiency was calculated on the basis of the information on the mass loss of concrete block after laser scabbling and the mass of collected debris in a container. The collecting efficiency was found to be over 85%.

Keywords: Concrete decontamination, Laser scabbling, Laser decontamination, Debris collector, Explosive spalling on concrete

Gamma Camera Applications of Radiological Characterization for Dismantling NPP

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Prior to dismantling a nuclear facility, full site characterization should be carried out to identify basic data for various stages of decommissioning, such as deregulation of sites and structures, selection of decontamination technology, decommissioning methods, and waste management and disposal. Radiological characterization is implemented through information collection, on-site measurement, sampling and analysis, and theoretical calculations and proven codes for radioactive material at the time of decommissioning of the nuclear facility. There are issues in that it takes a lot of time and money to collect and analyze samples for characterization of contaminated sites and radioactive structures. Therefore, in the entire process of decommissioning a nuclear facility, a technology that can quickly measure the radiological characteristics of various decommissioning objects and wastes on site is required. In this project, the utilization of gamma cameras that can be analyzed in the field for quick and accurate characteristic evaluation at the dismantling site was studied.

A gamma camera, iPIX from Canberra (Now it became Mirion Technology), was tested in this study. It is a unique gamma imager, which have a CdTe sensor with TIMEPIX chip and a coded aperture collimator, quickly locates and identifies low to high level radioactive sources from a distance while estimating the dose rate at the measurement point in real time. It also can be combined with CZT sensor which called iPIX-NID (nuclide Identification) provides users with clear understanding of radionuclides presence with no need of any spectroscopic knowledge. iPIX with iPIX-NID convert the gamma camera into a hot-spot detector with radionuclide information. To verify the applicability of a gamma camera in Nuclear power plant, it was implemented to Kori unit-1 which was permanently shut down from 2017.

Various Systems were observed at restricted area including reactor cooling system, boron recovery system, residual heat removal system, containment spray system, and etc. The locations of hot spots were clearly revealed by iPIX and these results can be used for selecting the locations of destructive samples and help to decide the conservative decision making. Condensate water systems in turbine building were also observed by a gamma camera and showed no nuclide. Based on this preliminary gamma camera applications, further investigation and tests will be carried out to Kori Unit-1.

Keywords: Gamma camera, iPIX, Kori, Characterization

Considerations on Preventive Actions of Critical Accidents From Operation of a Melting Facility for Radioactive Metal Waste

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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

Structural Stability of Recycled Concrete Waste

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The structural stability of the recycled concrete disposal container was evaluated and compared the applicability of the current design standards for recycled concrete of nuclear power plant. The structural stability requirement for concrete disposal containers is 37.7 MPa or more. As a result of the compressive strength test on recycled concrete, 50% of coarse aggregates of recycled concrete was 42.1 MPa. In addition, it was found that the bending strength and shear strength of recycled reinforced concrete beam exceeded the current design standard. Therefore, it is judged that recycled concrete containing coarse aggregates can be sufficiently utilized. It was possible to ensure the structural stability of the concrete container without changing the design specifications and reinforcing bars when recycled concrete is applied.

Keywords: Structural stability, Decommissioning, Nuclear power plant, Radioactive, Concrete waste, Recycled waste

An Approach of Process Development for the Project Management of Nuclear Power Plant Decommissioning

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The establishment of processes for the decommissioning a Nuclear Power Plant (NPP) is one of the objects that must be prepared in carrying out the decommissioning project. In particular, in the domestic situation, where there is no experience of decommissioning commercial NPPs, it is necessary to organize the tasks and contents well in advance for the successful initiation of the project. Therefore, this study intends to present a guide-level approach to develop management for domestic decommissioning projects.

As a documented template for recognizing a process, there may be a process map and description, and information such as the work structure and the relations between the activities should be indicated. In reality, activities will be managed through a set of computer system, so it would be better if the work content, activity flow, relation, management target information, computerization contents, etc. were materialized in the process. What is important here is to define the management areas and activities and draw the activity flow.

Domestically, it has rich experience in construction of NPPs and has a track record of exporting NPPs to the UAE. From these experiences, we have established a framework for standardized work in construction management and construction processes, and are performing them through a computerized system. Since the work of decommissioning has a similar nature to that of construction, we will be able to benchmark the procedure for the decommissioning from the construction management procedures. Typically, in the case of schedule management, the concept and structure of the construction process will be applicable to the decommissioning. Meanwhile, the licensee of domestic decommissioning is the same as the licensee that performs the operation, and the members who will perform the decommissioning also have experience working in the operation period. Therefore, the decommissioning works are an extension of the task during operation. Representatively, there are some processes that can be applied as it is even when decommissioning, such as dismantling work and the safety management process of the radiation zone.

Therefore, in carrying out the decommissioning of NPPs in Korea, processes and activities of the management area should be established from the construction processes with abundant experience and the processes during operation. Rather than making a completely new work process, this approach that properly reflects the existing work flow is expected to be an appropriate way to avoid the repulsion of employees and maladjustment to the new environment.

Keywords: Decommissioning, Processes, Activities, Project management

Characteristics of Radioactive Discharges and Anticipated Offsite Doses for Korean Nuclear Power Plants Under Permanent or Long-term Shutdown

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The International Atomic Energy Agency (IAEA) refers to the possibility of changes in the discharge characteristics of radioactive effluents that are different from those during operation when a nuclear power plants (NPPs) are decommissioned. In addition, the IAEA recommends differentiated radioactive effluent management for each phase during decommissioning that reflects changes in discharge characteristics, and changes to authorization and program that are different from those in operation.

Bonavito et., al. estimated the discharge and dose of liquid and gaseous radioactive effluents based on the decommissioning plan of the Trino NPP in Italy during decommissioning, but there is a fundamental limitation in that actual data were not used. Kang and Cheong analyzed the discharge characteristics of radioactive effluents at each activities of decommissioning after permanent shutdown using actual data on radioactive effluents from the United States and Europe, and performed theoretical modeling of discharge characteristics during permanent shutdown. However, there are limitations in that only the emitted radioactivity was considered, the dose assessment was not taken into account, and the improvement methods for the differentiated monitoring program for each phase of decommissioning mentioned in the IAEA were not proposed.

Most studies of radioactive effluents discharge from NPPs focus on normal operation, and studies of shutdown or decommissioned NPPs is very limited. Existing studies have not been extended to research on decommissioned NPPs, and there are limitations in that they do not consider the characteristics of decommissioned NPPs mentioned in the IAEA.

Therefore, this study aims to improve the effluent monitoring program based on the analysis of the discharge characteristics NPPs that are permanently or long-term shutdown and the change in offsite dose to public. For this purpose, research was conducted on Kori Unit 1 and Wolsong Unit 1 in Korea, which were virtually permanently shutdown, and other long-term shutdown NPPs due to prolonged planned outage maintenance or replacement/repair of equipment in nuclear facility. The discharge characteristics of each radionuclide group, and further, the effect of radioactive effluent released to the environment on the offsite dose are analyzed in details.

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Keywords: Nuclear power plants, Permanent shutdown, Long-term shutdown, Decommissioning, Radioactive effluents, Anticipated offsite dose

A Case Study on the Decontamination Reuse and Recycle of Contaminated Metal and Concrete for Decommissioning NPP

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Decontamination and Dismantlement (D&D) are of great interest to owner of decommissioning as a large number of old nuclear facilities around the world are either shutdown or soon to be decommissioned.

D&D are key steps in the decommissioning of nuclear power plants (NPPs). These activities typically generate a significant volume of radioactively contaminated waste. However, as much as 90% or more of this waste is lightly contaminated metal and concrete that could potentially be cleared for recycle or beneficial reuse, rather than disposed of as radioactive waste.

The objective of this study is to provide reference for the application of current technologies to cost-effectively reduce the volume of radioactive waste associated with decommissioning, through review of experiences with decontamination of NPPs materials for unrestricted release, recycle or reuse. Also, highlights the importance of ongoing efforts to harmonize regulations and standards for radioactive waste management globally to enable reuse and recycle of valuable materials generated during decommissioning.

The presented results in the balance of this study are organized to align with the sequence of operations for executing reuse or recycle of material for a decommissioning project.

Concrete from buildings has most commonly been used for backfill of voids onsite, while metal has most commonly been melted or cleared into the conventional scrap recycling industry. Copper and lead, commonly found in cables and shielding, have high residual value and are thus highly desirable for recycling.

Steel and stainless steel, while not inherently valuable, are present in many large components, such that decontamination for recycling can be cost-effective compared to disposal as radioactive waste.

The decontamination techniques range from simple, inexpensive methods to complex, aggressive methods, each with advantages in various scenarios and limitations in others.

Treatment often involves the sequential application of two or more decontamination techniques (e.g., chemical decontamination followed by abrasive blasting).

Strategies for the characterization of materials for recycling include analyzing material in place before dismantlement, analyzing removed samples before or after dismantlement, and evaluating bulk material removed after dismantlement.

If clearance and recycling are permitted, metals can be released to the conventional scrap recycling market, and concrete rubble can be used as backfill material onsite.

In general, successful reuse/recycle projects require consideration of reuse/recycling objectives and implementation of associated planning activities early in the decommissioning process.

The practicality of reuse/recycle depends on a number of high level (country and region-specific) and component level (material and case specific) factors.

Since this information is useful to those responsible for planning or implementing the decommissioning of nuclear facilities, it is expected that it will be of great help especially to those in charge of decommissioning plan and managers in charge of decommissioning projects.

Keywords: Decommissioning, Clearance, Reuse and Recycle

A Study on Long-term Integrity of Integrity of Polymer Concrete on Mechanical Stability

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The Korea Nuclear Safety Act defines a high integrity container (HIC) as “a radioactive waste packaging container that can maintain its integrity for more than 300 years under the general underground environment and disposal conditions in Korea”, and detailed technical standards are not described. The US Nuclear Safety Commission’s “Low-Level Waste Licensing Branch Technical Position on Waste Form” describes the detailed requirements for solidification and HIC. The main contents of the US NRC technical position include limiting the free standing water, minimum design life, demonstrating mechanical, thermal and radiation stability, etc.

In this study, the stability evaluation was performed to understand the mechanical strength with respect to horizontal and vertical loads. The basic property of polymer concrete was carefully evaluated, including compressive strength, structural fatigue resistance, etc. The long term creep test, loading of 40% of compressive strength, indicates that the polymer concrete exhibits good long term mechanical integrity.

Keywords: High integrity container (HIC), Polymer concrete (PC) mechanical stability, Compressive strength, Creep test

Evaluation of Integrity of HIC Through Evaluation of Chemical Resistance Over Time

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HIC refers to a radwaste packaging container that can maintain integrity for more than 300 years in the general underground environment and disposal conditions in Korea. For HIC, the integrity of containers is verified according to the HIC regulation guideline for LLW and ILW disposal. Existing material tests include mechanical stability, permeability resistance, corrosion properties, chemical durability and biological resistance.

In this study, a chemical durability test was conducted to prove the suitability of the HIC material by measuring the degree of chemical influence other than corrosion from the disposal environment. The chemical resistance evaluation method was used to simulate the disposal environment in the underground repository, and the amount of change in the physical properties of the degraded polymer concrete specimens according to the test time was confirmed. The technical standards considered leaching of material components, sulfation attack, acid attack, alkali, carbonate, and salt crystallization. The compressive strength and weight change of the specimens with time were checked by immersing them in a chemical solution that could leak major hazardous substances and wastes in the groundwater of the repository for several months. In addition, in order to evaluate the integrity in condition severe than the disposal environment, a flow was applied to a chemical solution having a concentration twice that of the basic chemical resistance test conditions, and the test period was extended twice to accelerate the deterioration of the specimen.

Keywords: High integrity container (HIC), Chemical resistance, Compressive strength

Experience Review on Decontamination of Concrete During Decommissioning of NPPs

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The permanent shutdown of NPPs (nuclear power plants) has been growing steadily around the world. Also, permanent shutdown of old NPPs has been determined following to Kori-1 and Wolsong-1 in the Korea. Among issues of decommissioning of NPPs, especially, the management of radioactive waste is the most sensitive issue. According to IAEA, a large volume of radioactive concrete waste would be generated from decommissioning of nuclear facilities. Also, EC (European commission) expected that about 500 million tons of concrete will be produced in Europe by 2060 due to decommissioning of NPPs. It is known that the radioactive concrete consists of surface contaminated concrete and activated concrete. So, if contaminants from radioactive concretes can be removed using specific technologies, volume reduction of concrete can be achieved.

Since there is no experience of decommissioning of NPPs in Korea, it is important to analyze previous cases. In this study, decontamination of radioactive concrete is analyzed through previous studies.

Decontamination technologies of concrete are composed of mechanical methods, chemical methods, and thermal methods. Mechanical methods are physical technologies separating contaminants from concrete using scabbling, milling, and vacuum cleaning. In chemical methods, contaminants were removed from concrete using an oxidizing agent/reducing agent, acid/base. Thermal methods are removal technologies using lasers, microwaves, and pulsed power discharge. Some methods still have practical use cases, and further research is needed on the issue of generation of secondary waste.

Review on the experience of decontamination of concrete show that waste of concrete generated during decommissioning of NPPs are expected to have effect of large volume reduction. However, many studies are needed because secondary waste and decontamination cost is sensitive issue of concrete generated during decommissioning of NPPs.

In order to successful decommissioning of NPPs in Korea, various research of decontamination of concrete are need

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Keywords: Radioactive waste, Decontamination, Decommissioning, Concrete

A Study on Activation Analysis of Activated Components: Sensitive Analysis on Impurity

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The permanent shut-down of Kori unit 1, the first nuclear power plant (NPP) in Korea, generates various radioactive waste. They are dry active waste (DAW), spent resin, concentrated waste, activated metals, etc. During normal operation of NPP, activated metals are rarely generated. The decommissioning of NPP, however, generates massive amount of metallic waste including activated metals and contaminated metals, while normal operation generates small amount of metallic waste.

The reactor vessel and internals are relatively highly activated components in NPP. Since they are exposed to the high concentration of neutrons during the operation, their contains relatively high radioactive nuclides. They activation analysis is usually performed to understand the radiological inventory of the activated reactor vessel and internal. The results offer various important information including, radiological inventory, waste classification, etc.

The impurities in the carbon steel and stainless steel have a great impact on the radiological inventory of the activated metals. The cobalt, nickel, niobium are primary elements that affects the activation analysis and waste classification. Especially, the cobalt, which transforms to ^{60}Co , plays an important role. The ^{60}Co , strong gamma emitter, affects the waste classification, safety analysis of decommissioning workers, and determination of segmentation and package plans.

In this paper, effects of impurity concentration on activation analysis is studied. The expected impurity from various sources, including NUREG/CR-3474, commercial NPP data, etc, and effects will be demonstrated. Also, the comparison between results and international experiences will be followed.

Keywords: Kori, Activation analysis, RV, RVI, Impurity

Long-term Stability Analysis on Polymer Concrete for High Integrity Container Using Cl Ion Diffusion Test

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During the normal operation boron concentrates and spent resins are generated. The boron concentrates are treated by concentrated waste drying system (CWDS) and results in fine powder form. The solidification or application of high integrity container (HIC) is required for the disposal of the dried boron concentrates. The spent resin is stored in storage tank after the water treatment. The spent resin also requires solidification or application of HIC to satisfy the waste acceptance criteria (WAC) in Korea. The solidification process requires periodic validation. The repeated validation and complicated process hesitates the practical application.

The application of HIC offers various advantages, including flexible free standing water requirement, higher waste loading compared to solidification, and simple process. The polymer concrete (PC), which is a primary component for PC-HIC exhibits good material stability. The expected transportation mechanism of nuclide in the PC-HIC are 1) diffusion by concentration, 2) permeation by pressure, and 3) capillary suction when considering the disposal condition. Since the PC-HIC effectively prevents the intrusion of neighboring water and volume of free standing water is lower than 1%, it seems that diffusion by concentration is the primary transportation mechanism.

In this study, the property of PC is investigated based on Cl ion diffusion test to evaluate the material reliability. The results indicate that PC exhibits superior stability compared to ordinary portland cement. In addition, the reliable life time of PC is estimated base on the element transportation phenomena.

Keywords: High integrity container, Spent resin, Boron concentrates, Polymer concrete

Management Lessons Learned Gained From the US Three NPPs Decommissioning Cases

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Numerous nuclear power plants that had been built in the late 20th century have entered the aging phase and are scheduled to be decommissioned. The decommissioning project of a commercial nuclear power plant is an array of complex processes involving the activities of site characterization, decontamination, dismantling, and site restoration. Hence, a number of essential factors, such as scheduling, work progress, and staffing, should be taken into account while the decommissioning plan is drafted and modified. Guidances on managerial and social aspects of decommissioning have been rare as compared to those of technical viewpoints. Nonetheless, the nuclear industry in the US has presented no little amount of experience on their decommissioning projects dealing with those perspectives. Thus, three sets of the case study were conducted to obtain useful lessons learned. The Maine Yankee nuclear power plant initially acquired 40 years of the operating license, it was in operation for only 25 years from 1972 until 1996. The owner group decided to shut down because of the deterioration of the profitability in 1997. The case of the Maine Yankee project enlightened the importance of the contract management and stakeholder relations. The Rancho Seco nuclear power plant is a single-unit nuclear power reactor site with 913 MWe output that commenced commercial operation in 1975. The Rancho Seco that had become the first-ever reactor shut down by a public voting introduced several innovative approaches for the decommissioning, some of which turned out to be very successful. The SONGS 1 commenced the commercial operation in 1968 and had been decided to cease its operation permanently due to a steep decline in profitability in 1992. The SONGS 1 presented worthwhile lessons in terms of project management. In this study, several lessons learned related on managerial, engineering, and regulatory/social aspects considered during the NPP decommissioning will be reviewed and discussed.

Keywords: Decommissioning, Maine yankee, Rancho seco, SONGS

Contaminated Water Management Prospects on the Fukushima Daiichi Nuclear Power Plant Accident

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The Fukushima nuclear power plant accident, which was caused by the Great East Japan Earthquake on March 11, 2011, is of great concern to the Korean people. The scope of interest is wide and diverse, from the nuclear accident itself and the damage situation, to the current situation in Fukushima Prefecture and Japan, and to the safety of Japanese agricultural and fishery products. Concerns about nuclear safety following the Fukushima nuclear accident have a significant impact on neighboring nation's energy policy.

It has been 11 years since the Fukushima nuclear accident. In neighboring nation society, the nature and extent of damage caused by the Fukushima nuclear accident, the feasibility of follow-up measures at home and abroad, the impact on neighboring nations, and the direction of nuclear policy reflecting the lessons of the accident are hotly debated topics. Recently, the controversy has grown further as it is intertwined with Japan's concerns about the safety and discharge of the contaminated water into the sea, and conflicts over domestic nuclear power policies. About 1.29 million tons, as of March 24, 2022, of the contaminated water are generated, which is close to the 1.37 million tons of water storage capacity. In response, the Japanese government announced on April 13, 2021, that it plans to discharge the contaminated water into the sea from 2023. This study evaluates the amount of the contaminated water that has passed through the ALPS and reviews the preparations and related facilities for ocean discharge after diluting the contaminated water. In addition, it is intended to forecast the various impacts of ocean discharge.

Keywords: Fukushima NPP, Contaminated water, ALPS, Ocean discharge

A Study on International Experiences on Concrete Characterization of Decommissioned Nuclear Power Plants

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During the decommissioning of nuclear power plant (NPP), massive amount of concrete wastes is generated, which are non-radioactive and radioactive. The concrete is a representative construction material which affords reliable structural stability, good formability, and trustful integrity. Also, its reasonable neutron absorbing property allows the various application for many components, including building construction material, bio-shield concrete, etc. Due to the noted aspects of concrete, the radiological concrete characterization is classified as an important process for development of effective strategy for concrete management, in terms of process management and financial control during the decommissioning.

The characterization of bio-shield concrete is important in waste management. The understanding and characterization of activation depth is essential for the determination of waste management strategy, classification of bio-shield concrete, and process development of decommissioning. On the other hand, concrete for construction application requires the evaluation of surface contamination of them. The concrete for containment building and its structure is rarely activated, but surface contaminated.

In this paper, the reactor data from representative PWR reactors in the US is studied. The experience on Yankee Rowe, Maine Yankee, and Connecticut Yankee NPPs are systematically studied. The Yankee Rowe was a 4-loop PWR of Westinghouse design with 185 MWe. The Main Yankee was a 3-loop PWR of Combustion Engineering design with 864 MWe. The Connecticut Yankee was a 4-loop Westinghouse type with 560 MWe. The characterization studies on bio-shield concrete will be discussed in this paper, including activation depth, primary nuclides, etc.

Keywords: Bio-shield concrete, Decommissioning, Radioactive waste, NPP

Review on Concrete Waste Management Technology for Decommissioning of Nuclear Power Plant

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The decommissioning of nuclear power plant (NPP) generates large amount of waste. Since the most of the concretes are slightly surface contaminated, the accurate characterization and region-specific surface decontamination are important for the efficient waste management. After the effective surface decontamination and separation, most of the concrete waste from decommissioning of NPP can be classified as a clearance waste.

Various surface characterization and decontamination technologies are suggested. The mechanical technologies are simple and offers direct application. The laser-based technologies offer efficient separation and surface contamination. The high price, however, hesitates the application of the process. The nitro-jet technology, which is based on the evaporation of liquid nitrogen, allows the effective decontamination. However, the high price and uncertainty of large are application hinders the practical application in NPP decommissioning.

In this paper, various technologies for characterization, handling, treatment, etc., will be discussed. The advantages and disadvantages of the technologies will be discussed, in terms of practical applications.

Keywords: Decommissioning, NPP, Nitrojet, Concrete

A Study on Radioactive Waste Level of the Calandria and Concrete Structures for Wolsong Unit 1

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Wolsong unit 1 (W1), which is a CANDU-6 type PHWRs that had been operated for 30 years since 1983, was shutdown in 2019. In this study, the radioactive waste levels of calandria and concrete structures were calculated to establish a decommissioning plan for W1. The specific systems within the scope of this study were grouped into 6 major categories as follows: Calandria, End Shield, Fuel Channel Assembly, Reactivity Control Device, End Shield Support, Vault.

The main operating history of W1 is that the re-tubing project was performed. These characteristics and operation history were reflected in the evaluation. The neutron flux and energy spectrum of each structure were calculated by using MCNP code, and ORIGEN code is implemented to the calculation of radioactivity for each nuclide using the results from MCNP and the material information of the structure. As for the impurity information, ASTM B350, B351, B353 standard was used for zircaloy alloy. For other alloy, impurity information provided by NUREG/CR-3474 was applied. Since W1 is expected to be decommissioned immediately, the waste level was evaluated under cooling conditions for 5 years after permanent shutdown.

Through the level evaluation of each component obtained as a result of the study, it can be used as basic data for the radioactive waste management of the decommissioning plan.

Keywords: Decommissioning plan, CANDU6, Calandria, Radioactive waste

A Preliminary Study on Facilitation of Decommissioning of Cyclotron Facilities Considering Neutron-Activated Waste Generation Using MCNP and FISPACT-II

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In worldwide, tens of thousands of units of particle accelerators have been used and more than 97% of those accelerators are used for dedicated medical of commercial applications. Radionuclide production cyclotron produce several positron-emitting radionuclides such as ^{18}F by $^{18}\text{O}(p,n)^{18}\text{F}$ reaction which generates secondary neutrons. It is of note that these neutrons cause neutron activation in structures and components of cyclotron facilities. Therefore, International Atomic Energy Agency had addressed that a well-developed estimate of the neutron activation induced radioactive inventory of accelerator facilities is needed for the proper planning and safe implementation of decommissioning using proven methods or codes that can be used to perform activation calculations. Moreover, IAEA suggested that during the operation of cyclotrons, concrete walls become radioactive over time and this radioactivity needs to be fully characterized as part of early decommissioning planning. In this study, Neutron activation in the medical cyclotron facilities was evaluated with the MCNP and FISPACT-II code to analyze the generation of decommissioning radioactive wastes during facilities dismantling.

For the reference case, residual radioactivity concentration of each activation product (e.g. ^{60}Co , ^{152}Eu , etc.) was calculated and the sum of fractions of the activity concentration of each radionuclide divided by its clearance level was exceeded 1.0 at each calculation point which means radioactive waste generations during decommissioning of the facility. Several points show the calculated sum of fractions (SoF) at inside wall were bigger than the surface wall. The reason of these phenomena is that the slowdown of the incident neutron energy at the inside wall due to neutron attenuation and larger thermal neutron flux than surface wall. It is of note that each activation reaction cross-section was dominant at thermal neutron energy band. Sensitivity analysis was conducted to analyze the effects of design characteristics (e.g. beam energy and current, operation period, and workload). The SoF was exceeded 1.0 at the least activation condition (i.e. 9 MeV, 10 μA) if the operation period was 10 years. For the realistic condition such as 13 MeV, only 10 μA of beam current case shows the SoF was under union. On the other hand, 19 MeV, 60 μA , and 10 years operation case shows the SoF as 20.4 which means the clearance rule can be applied only after 21 years of decay-in-storage.

The result of this study can be used for proper planning of decommissioning and/or new installation of cyclotron facilities include considerations of radioactive waste management.

Keywords: Neutron activation, Radioactive waste management, Decommissioning, MCNP, Monte-carlo

International Experience Review on Sampling for Radiological Analysis

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In Korea, Starting with the permanent shutdown of Kori Unit1, decommissioning of commercial nuclear power plant is underway. Although various technologies are required to decommissioning a nuclear power plant, the most important technology is the characterization of radioactive waste. In particular, it is possible to establish an accurate decommissioning plan and estimate cost for radioactive waste through accurate characterization of reactor vessel (RV), reactor vessel internal (RVI) and bioshield, which are highly activated waste.

In Slovakia's V1 nuclear power plant, two units were shutdown in 2006 and 2008, respectively, and decommissioning license was approved in July 2011. Before approving the decommissioning license, the decommissioning database project was carried out from 2008 to 2011. At this time, radioactive evaluation was performed through sampling and radiological analysis of radioactive structures.

Keywords: Decommissioning, Radwaste, Characterization, Sampling

An Evaluation of Worker Radiation Exposure During Dismantling for Reactor Vessel in Kori Unit 1

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This study is for evaluation and optimization of workers' radiation exposure for dismantling Reactor Vessel (RV) at Kori unit 1 in connection with its decommissioning process for the purpose of establishing Radiation Safety Management Plan. This is because the safety of workers in a radiation environment is an important issue. The basis of radiological conditions of this evaluation is supposed to be those of 10 years after the permanent shutdown of Kori unit 1 when dismantling work of Reactor Vessel would suppose to be started. Dose rates of work areas were evaluated on the basis of spatial dose rate derived from activation level calculated by MCNP (Monte Carlo N-Particle Transport) and ORIGEN-S code. RV are radiated by neutrons during operation, creating an environment in which it is difficult for operators to access and work. Therefore, the RV must be dismantled remotely. However, due to work such as installing devices or dismantling surrounding structures, it is not possible to completely block the access of workers. Accordingly, the exposure of workers to the RV dismantling process should be assessed and safety management carried out. The dismantling process of Kori unit 1 RV was developed based on in-situ execution in atmospheric environment using the oxygen-propane cutting technology as the following steps of Preparation, Dismantling of Peripheral Structures, Dismantling of RV and Finishing Work.

For evaluation of exposure of RV dismantling work, those processes of each steps are correlated with spatial dose rates of each work areas where the jobs being done. Results of the evaluation show that workers' collective dose for RV dismantling work would be in the range of 536–778 man- mSv. The most critical process would be dismantling of upper connecting parts of RV with 170–256 man-mSv while among the working groups, the expert group performing dismantling of ICI (In-core instrumentation) nozzles and handling & packaging of cut-off pieces is evaluated as the most significantly affected group with 37.5–39.4 man- mSv. Based on the evaluation, improvement plan for better working conditions of the most critical process and the most affected workers in terms of radiation safety were suggested.

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Keywords: Dismantling, Reactor vessel, Radiation dose, Safety, Management

Key Elements of Radiological Characterization in Transition Period for Decommissioning of a Nuclear Facility

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Radiological characterization, one of the key factors for any successful decommissioning project for a nuclear facility, is defined as a systematic identification of the types, quantities, forms, and locations of radioactive contamination within a facility. This characterization is an essential early step in the development of a decommissioning plan, in particular during transition period after permanent shutdown of the facility, and also to be used for classification of decommissioned radioactive wastes so that their disposal criteria can be met. Therefore, the characterization should be well planned and performed.

In the transition period, the characterization information developed during the operational phase is usually reexamined with respect to the applied assumptions, the actual status of the facility after shutdown, the accuracy of the required measurements and changes in its radiological properties to support the development of the final decommissioning plan. Based on some national (Korean, USA's and Japanese) laws including the related regulations, and some related documents published by OECD/NEA, IAEA, and ASTM, key elements of radiological characterization, which should be developed in the transition period, could be proposed as the followings.

The key elements might be an operational history including facility operation history and contamination by events and/or accidents, radiological inventory of the facility and site area, characterization survey including in-situ survey and/or sampling and analyses, radiological mapping (which is able to identify radiological contamination levels of SSCs, and the facility area and, if contamination may be suspected, the surroundings) with tabulating, residual radioactivity (or derived concentration guideline levels) of selected major radionuclides for remediation of the site, (retainable and retrievable) recording, and quality control and quality assurance. In review process of the operational history, interviews of current or former long-tenured knowledgeable employees of the facility should be conducted to identify conditions that may have been missing from the records.

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Keywords: Radiological characterization, Transition period, Decommissioning, Radiological mapping, Residual radioactivity

Management Plan for Liquid Radioactive Waste in the Waste Treatment Facility

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Starting with the permanent shutdown of Kori Unit 1, the first waste treatment facility in Korea will be built on the Kori site. In this facility, major process such as decontamination, cutting, radiation measurement and volume reduction of decommissioning waste are performed, and radioactive liquid waste is generated by the waste treatment process and personnel decontamination. The generated liquid waste is finally discharged to the sea through radioactive monitoring system after sufficient treatment to meet the standard radiological effluent control. Whereas the treated liquid waste is additionally diluted through the circulation water discharge conduit and discharged to the sea in the operating nuclear power plants, there is no circulation water in the waste treatment facility. Therefore, a new discharging method for dilution after treatment should be considered. In this paper, the treatment concept and discharge method of radioactive liquid waste system in waste treatment facility are reviewed.

Keywords: Waste treatment facility, Liquid waste, Treatment, Discharge

Preliminary Study on Potential Impact of Surrounding Structures to Neutron Activation Distribution in Medical Cyclotrons Using MCNP With FISPACT Code

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For producing radionuclides which were mostly used in medical purposes, for instance, Positron Emission Tomography (PET), there were about 1,200 PET cyclotrons operated in 95 countries based upon IAEA database (2020). Besides, including PET cyclotrons, demands for particle accelerators are continuously increasing. In Korea, about 40 PET cyclotrons are in operating phases (2020). Considering design lifetime (about 30–40 years) of cyclotrons, there will be demands for decommissioning cyclotron facilities in the near future.

PET cyclotron produces radionuclides by irradiating charged particles to the targets. During this phase, nuclear reactions ($^{18}\text{O}(p,n)^{18}\text{F}$, $^{14}\text{N}(d,n)^{15}\text{O}$ etc.) produce secondary neutrons which induce neutron activation of accelerator itself as well as surrounding infrastructures (the ancillary subsystems, peripheral equipment, concrete walls etc.). Most of the ancillary systems including peripheral equipment can be neutron activated, since, most of them were made of steels. Steels like stainless steel or carbon steel may contain some impurities, typically cobalt. Although, there were several researches evaluating activation of concrete walls and accelerator components, estimating the activation and influence on neutron interaction of the other surrounding infrastructures were insufficient.

In this study, by using computational neutron transport code (MCNP 6.2), and source term calculation code (FISPACT- II), we estimated neutron distribution in cyclotron vault and activation of ancillary subsystems including some peripheral equipment. Also, using Au foil and Cd cover, we measured thermal neutron distribution at 16 points on the concrete wall, and compared it to calculated results (MCNP). Even though, the compared results matches well, there was a discrepancy of neutron distributions between presence and absence of those equipment. Additionally, in estimating activation distributions by calculating, most of the steel-based subsystems including peripheral equipment should be managed by radioactive wastes after 20 years of operation.

Throughout this study, we could find that influence on neutron interaction of those equipment can affect neutron distribution in concrete walls. This results vary the activation depth as well as location of the hot contaminated spot in concrete wall. For estimating or evaluating activation distributions in cyclotron facilities, there was need to consider some equipment located in cyclotron vault.

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Keywords: Cyclotron, Decommissioning, Neutron activation, Neutron distribution, MCNP, FISPACT

A Study on Optimization of Radiation Shielding for Radioactive Waste Container Using DEMplus

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As the decommissioning of nuclear power plants increases, there is an increasing interest in the amounts of radioactive waste. Especially, the radiation dose limit for packaging of radioactive wastes shall not exceed $2 \text{ mSv} \cdot \text{h}^{-1}$ and $0.1 \text{ mSv} \cdot \text{h}^{-1}$ on contact and at 2 m, respectively in South Korea. The DEMplus provides various environmental geometry and all properties such as materials, absorptions, and reflections and the estimation of the radiation dose rates is based on the radiation interactions of the designed 3D geometry model. With the consideration of the radiation dose rate by using DEMplus and its strategy of packaging plan, the radiation shielding was optimized and estimated in this paper. The modular shielded containers (MSC) with shielding inserted were used for radioactive wastes that require shielded packaging. In order to verify the accuracy of the estimated radiation dose rate by using DEMplus, the estimated results were compared with those obtained using MicroShield. The trends of the estimated radiation dose rates using DEMplus and the estimation of MicroShield were similar to each other. The results of this study demonstrated the feasibility of using DEMplus as a means of estimating the radiation dose limit in packaging plan of the radioactive waste.

Keywords: Decommissioning, DEMplus, MicroShield, Modular shielding containers (MSC), Radioactive waste

Case Study for the Preparation of Methods for Demolition and Disposal of Contaminated Areas in Korea Research Reactor 1&2

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Korea Research Reactor 1&2 (KRR-1&2), Korea's first research reactor, began dismantling in 1997. As of 2022, the demolition of general areas such as offices has been completed, and contaminated areas such as reactor rooms remain. On the other hand, construction waste generated in contaminated areas of nuclear facilities cannot be disposed of as general industrial waste. It is predicted that about 5,000 tons of construction waste will be generated if the contaminated area of KRR-1&2 is demolished. In this study, the application plan for the demolition of contaminated area of KRR-1&2 was reviewed through a review of laws and cases related to domestic and overseas disposal.

The only method for disposing of construction waste in contaminated areas that can be applied in Korea is clearance in accordance with Nuclear Safety Commission Notice No. 2020-06. In addition, there has been no case of demolishing large-scale nuclear facilities in Korea. Therefore, there are limitations in domestic laws and standards to be applied to the dismantling of contaminated areas of KRR-1&2. The IAEA and the United States specify comprehensive matters such as optimization of radiation protection and minimization of waste products. The EU recommends demolition after decontamination by removing contaminated areas before demolition of buildings. It also presents three options for reuse, recycling, and disposal of buildings and building waste. In particular, in the case of Germany, detailed radioactivity measurement methods for deregulation of buildings and building waste are presented in accordance with the EU's guidelines. As a result of synthesizing this, it is judged that the EU and Germany building clearance plan will be suitable for domestic application.

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Keywords: KRR-1&2, Decommissioning, Radioactive waste, Disposal

Economic and Green Recovery of Metals From Deep Eutectic Solvents Utilized for Radioactive Metal Waste Decontamination

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Decontamination of spent nuclear fuel from decommissioned nuclear reactors is crucial to reduce the volume of intermediate-level waste. Fuel cladding hulls are one of the important parts due to high radioactivity. Their decontamination could possibly enable reclassification as low-level waste. Fuel cladding hulls used in research reactors and being developed for conventional light water reactors are Al-Mg and Fe-Cr-Al alloys, respectively. Therefore, the recovery of these component metals after decontamination is necessary to reduce the volume of highly radioactive waste. Electrochemical approach is often chosen due to its simplicity and effectiveness. Non-aqueous solvents, such as molten salts (MSs) and ionic liquids (ILs), are preferred to aqueous solvents due to the absence of hydrogen evolution. However, MSs and ILs are limited by high temperature and high synthesis cost, along with toxicity issues.

Deep eutectic solvents (DESs) are synthesized from a hydrogen bond acceptor (HBA) and donor (HBD) and exhibit outstanding metal salt solubility, wide electrochemical window, good biocompatibility, and economic production process. These characteristics make DES an attractive candidate solvent for economic, green, and efficient electrodeposition compared with aqueous solvents such as acids or non-aqueous solvents such as MSs or ILs.

In this research, the feasibility of electrodeposition of Al-Mg and Fe-Cr-Al alloys in ChCl:EG, the most common DES synthesized from choline chloride (ChCl) and ethylene glycol (EG), will be tested. A standard three-electrode electrochemical cell with an Au plated working electrode and Al wires for counter and reference electrodes is utilized. Two electrolyte solutions (Al-Mg and Fe-Cr-Al) are prepared by dissolving 100 mM of each anhydrous metal chloride salts (AlCl₃, MgCl₂, CrCl₃, and FeCl₂) in ChCl:EG. Cyclic voltammogram (CV) is measured at 5, 10, 15, and 20 mV·s⁻¹ to observe the redox reactions occurring in the solutions. Electrodeposition of each alloy is performed via chronoamperometry at observed reduction potentials from CV measurements. The deposited surfaces and cross-sections are examined by scanning electron microscopy and energy dispersive spectroscopy (SEM-EDS) to analyze the surface morphology, cross-section composition, and thickness. Authors anticipate that the presence of different metals will greatly affect the possibility of electrodeposition. It is expected that although all metals are distributed throughout the surface, the morphology, in terms of particle size and shape, would differ depending on metals. Different metals will be deposited by layers of an approximate thickness of a few μm each. This research will illustrate a potential for recovery and electrodeposition of other precious radioactive metals from DES.

Acknowledgements

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Keywords: Green chemistry, Metal recovery, Metal electrodeposition, Cyclic voltammetry, Nuclear waste management

Study on the Effect of Pressurizer Cutting Scenario

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When a permanently-closed nuclear power plant is to be decommissioned, large structures targeted to be cut in the process include a steam generator, reactor, and reactor coolant pump (RCP). Although there are sufficient preliminary studies being done on these structures to assess the radiation exposure dose, relatively fewer studies are underway regarding pressurizers. Therefore, preliminary evaluations are required to prevent workers from being overexposed to radiation coming from a pressurizer and to avoid an unnecessary increase in the decommissioning cost.

This study created a cutting scenario based on disposal drums for solid radioactive wastes. The cutting scenario was based on 200-liter and 320-liter drums for solid wastes and on the assumption that all cutting operations were done 100 centimeters away from the structure to be cut. When the cutting process of a Pressurizer is carried out per scenario, the 200-liter drum produces 272 pieces, whereas the 320-liter counterpart generates 234 pieces. Given that South Korea allocates 75,550 KRW per liter (based on 200 L) for the disposal cost, an increase in the number of drums leads to an exponential growth of the decommissioning cost, which fuels the need to establish more organized cutting strategies.

Meanwhile, in terms of radiation dose, plasma, laser, and flame cutting techniques were estimated to record 0.232 mSv, 0.299 mSv, and 0.213 mSv respectively for 200 L, and 0.195 mSv, 0.251 mSv, and 0.179 mSv respectively for 320 L (based on DF-90). When compared with the annual dose limit of 100 mSv ($0.0057 \text{ mSv}\cdot\text{hr}^{-1}$), the above numbers registered for both 200 L and 320 L were estimated to satisfy the dose limit, with only a negligible difference in the dose between the two capacities.

The results generated from this study are expected to be utilized as a meaningful basis to identify applicable cutting techniques of a pressurizer as part of the decommissioning operation and to establish its cutting plans in compliance with ALARA.

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Keywords: Pressurizer, Cutting scenario, ALARA, Decommissioning, External exposure

Highly Selective Cesium Capture Under Acidic Conditions After Soil Washing

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A large amount of acidic wastewater was generated from the soil washing process. This study focuses on the capture for the radionuclide, especially cesium (Cs), in soil washing wastewater. We conducted the two-step process to reduce the amount of radioactive wastewater after soil washing using the coagulants and adsorbents in each step. We synthesized the adsorbent to capture Cs radionuclides in acidic wastewater ($\text{pH} < 1$). Based on the results, we found that the optimum ratio (NiFC: PAN) was 3.5:1 for the removal efficiency and strength of adsorbent. We compare the NiFC powder and NiFC-PAN for removal efficiency and separation of adsorbent after batch test, showing that the removal efficiency and separation of NiFC-PAN was lower and higher than NiFC power, respectively. We conducted the radioactive experiment to evaluate the satisfaction below release criteria ($< 10 \text{ Bq} \cdot \text{L}^{-1}$, Cs), reporting that NiFC-PAN adsorbent meet the release criteria. These results showed that NiFC-PAN is promising adsorbent for Cs capture in strong acidic wastewater generated from soil washing and separation after capture process.

Keywords: Cesium, Radionuclide, Soil, Wastewater, Strong acidic, Adsorbent

Quantitative Evaluation of Vinyl and Styrene Polymers Performance for Detritiation of Low-level Tritiated Water

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Detritiation of low-level tritiated water has become global issue after Fukushima accident. Several attempts have been made to reduce the radioactivity of Fukushima tritiated water below legal limit of nuclear plant effluents ($\sim 10^4$ Bq·L⁻¹). Various technologies such as water distillation, electrolysis, and catalytic exchange were tested to treat the tritiated water, however, those demand enormous expense to achieve the goal due to low process efficiency. It highlights that the performance enhancement of current technologies is necessary to improve economic feasibility.

We have quantitatively evaluated the separation performance of various polymers toward low-level tritium ($\sim 10^5$ Bq·L⁻¹) through batch experiments. The polystyrene with grafted by 20 types of functional groups (Tris (2-aminoethyl) amine, dimethylaminomethyl, isocyanate, mercaptomethyl, aminomethyl, hydroxymethyl, triphenylphosphine, morpholine, 2-chlorotriyl amine, 4-(dimethylamino) pyridine, poly (vinyl chloride) carboxylated, poly (4-vinyl pyridine), p-toluenesulfonic acid, p-toluenesulfonyl hydrazide, piperidine, acetyl polystyrene, 2-chlorotriyl chloride, piperazine, diethylene triamine, poly (vinyl chloride)) were suspended in HTO solution (initial activity = 4.7×10^5 Bq·L⁻¹). After the equilibration, the suspension was filtered using 3 kDa membrane filter and the activity in filtrates were quantified by LSC (HIDEX-300 SL). The results demonstrate the detritiation efficiency and separation factors of functional groups toward tritium. Carboxylic group (COOH) showed the most reactive performance as detritiation efficiency of $\sim 4\%$. Compared to other functional groups, styrene sulfonyl groups including sulfonyl amide (SNH₂) and sulfonyl hydrazide (SNHNH₂) revealed promising performance for tritium separation as separation factor of 10.97 and 3.85, respectively. However, sulfonyl hydroxide (SOH) which is known as reactive functional group to tritium exchange showed the poor performance (detritiation efficiency: 0.68%; separation factor: 3.02).

This study could suggest the promising functional groups for detritiation of low-level tritiated water which can be utilized to enhance the performance of current technologies. For example, reactive functional groups can be grafted on the surface of packing material within distillation tower resulting in the increasing detritiation efficiency.

Keywords: Tritium, Functional groups, Separation factor, Detritiation, Low-level tritiated water

A Study on the Design Base Accidents Scenario of Non-Reactor Facilities

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The decommissioning of Kori unit 1 is just around the corner. Accordingly, it is required to construct a hot cell facility for decommissioning nuclear power plants to analyze the characteristics of intermediate-level waste and low-level waste generated in the decommissioning process. In this study, a Design Base Accident (DBA) scenario of the facility is developed. To identify and characterize potential hazards at the facility, a Preliminary Fact Sheet (PFS) is filled out and consider external events in consideration of the surrounding site environment. The external event screening and evaluation method is based on the external event evaluation method covered in the probabilistic risk assessment. In PFS, only natural and artificial hazards that may have a meaningful impact on the facility are considered as the sources of the accident, and accident prevention and mitigation systems, etc., which exist in each compartment or facility, are described. Based on PFS and external events, potential hazard assessment is systematically performed using each potential hazards, impact and defense function identified using the preliminary hazard analysis (PHA) methodology. The potential hazard analysis methodology applied to this assessment is a qualitative assessment method consistent with the US DOE Hazard Analysis methodology (DOE 1992b; DOE 1994b). After that, the potential mitigation functions that can be used under normal, abnormal and accident conditions are examined, and the contribution of public and workers to safety is evaluated. The results of the PHA are basic data that prioritize potential hazards and can be used to develop potential accident scenarios. Among potential hazards generally considered for non-reactor facilities, only possible accidents during operation of the facilities are selected as potential hazards. The level of potential hazards is obtained by qualitatively examining the frequency and consequence estimates for each hazard or accident scenario developed in PHA. Based on the results of the potential hazards assessment, representative accidents that require further quantitative analysis are screened. Selected accidents are DBA and are the most dangerous and most significant impacts on workers.

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Keywords: Design base accident, Preliminary hazard analysis, Preliminary fact sheet, Non-reactor facility

A Review on the WDF Application Methods and Their Characteristics

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The decommissioning cost of a nuclear power plant (NPP) is largely composed of activity-dependent costs, period-dependent costs, and collateral costs. And activity-dependent costs for each decommissioning activity are composed of five cost elements: Removal, decontamination, packaging, shipping and disposal. Among these, the removal cost elements are calculated by multiplying the appropriate inventory data element by the corresponding unit factors (UF), which are developed in terms of labor hours to perform an activity on a per unit basis.

The labor hours included in UF is calculated under theoretical working conditions, which, after being multiplied by labor rates, composes unit cost factor (UCF) along with material cost. In the actual working conditions, there are number of factors that increase the time needed for performing a task. The effects of these factors are taken into consideration by means of work difficulty factor (WDF), expressed as a percentage of increase of the working time, comparing to an unimpeded working situation. WDF, by increasing the labor hours and consequential labor cost in UCF, makes it possible to calculate the actual removal cost. There are about five types of adjustment factors commonly used as WDF: Height, Respiratory protection, Radiation, Protective clothing, Work break.

Considering the different working conditions, all of the five factors' combination could be used theoretically, which results in the huge increase of the number of WDFs. For practical purpose, two representative WDF application methods has been used in the dismantling decommissioning cost evaluation program: A separate development of the UCFs, WDFs applied to the decommissioning area. In the first method, all of the UCFs, having different working environment, should be developed separately by the cost estimator. In the second method, UCFs are to be allocated to the relevant decommissioning areas where WDF sets are predefined by the cost estimator.

In this study, the components of the decommissioning cost, the relation between UCF and WDF, and WDF application methods were reviewed. The result of review implies that WDF has a great influence on decommissioning cost. Additionally, since WDF application methods have somewhat limitations and complexity, their characteristics should be sufficiently examined by the user before being used.

Keywords: Decommissioning cost, Unit factor, UCF, WDF

Behaviors of Metals During the Cyclic SP-HyBRID Process of Reactor Coolant System in a Pressurized Water Reactor

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Activated corrosion products deposited on the reactor coolant system in a nuclear power plant should be removed to reduce the radiation exposure to workers. Chemical decontamination processes using organic acids have been widely applied to remove the activated corrosion products. However, they are highly corrosive to the base metal and generate a considerable amount of ion exchange resin waste, which is hard to be treated. In order to resolve this problem, KAERI has been developed a chemical decontamination process using chelate-free inorganic acid, HyBRID (Hydrazine Based Reductive metal Ion Decontamination) process. Especially, the Cyclic SP (Sulfuric acid/Permanganate)-HyBRID process was suggested as the decontamination process for applying to the remove the double oxide layer generated on the reactor coolant system in the pressurized water reactor (PWR). During the Cyclic SP-HyBRID process, the process is continuously applied without discharging or recharging of the decontamination process solution from the primary circuit. Thus, it is necessary to include the removal processes of the decontamination reagents middle of the Cyclic SP-HyBRID process, e.g., 'Mn removal step' for removing the permanganate ions and 'hydrazine decomposition step' for decomposition of the remaining hydrazine. During these removal processes, the metal ions can also be removed from the process solution. In this study, the behaviors of metals were investigated during the Cyclic SP-HyBRID process. The concentration changes of metal ions in the process solution were analyzed using atomic absorption (AA) spectroscopy. The metal precipitates generated during the process were characterized using X-ray diffraction (XRD) and Fourier Transform Infrared (FT-IR) spectroscopy. From the results of the analysis, it was observed that the metal ions dissolved in the process solution were converted into metal hydroxides and precipitated at the Mn removal process. It was confirmed by equilibrium calculation result that the OH⁻ ions generated at the Mn removal can react with the metal ions and form the metal hydroxides. It is considered that this removal behaviors of the metals can contribute the decontamination performance.

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Keywords: Decontamination, HyBRID, Cyclic SP-HyBRID, Metal behavior

Evaluation of Physicochemical Characteristics of Radioactive Sulfate-rich Powder Waste Form Immobilized by Low Temperature Sintering

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Sulfate-rich waste powder containing a radioactive nuclide is generated from chemical decontamination process and radioactive liquid waste treatment using ion exchange resin. The radioactive sulfate-rich waste powder should be stabilized for final disposal. The techniques for immobilization of the radioactive sulfate-rich waste powder such as hydraulic cement, geopolymer, and iron phosphate glass have been applied, however, there are limitation in these techniques. Firstly, the hydraulic cement cannot applied to the wastes containing high concentration of sulfate because the expansion, cracks, and disintegration can be happened in the waste form. Geopolymer has a low density although they can be used as a good binder. The iron phosphate glass can be utilized, however, a considerable amount of SO₂ gas is emitted due to the high sintering temperature. In this study, immobilization of radioactive sulfate-rich waste powder was carried out to resolve above problems by applying low temperature sintering method using a low-melting glass. As a result, it was confirmed that the waste form has a high bulk density. The compressive strength of the waste form was over 40 MPa, which is higher than the acceptance criteria (≥ 3.44 MPa). From ANS 16.1 test, it was verified that the waste form met the acceptance criteria of the leachability index (≥ 6). It was also confirmed that the waste form was chemically durable through product consistency test (PCT). In addition, the chemical stabilities of waste forms were compared following the sintering condition and the composition of the waste forms. The difference of the chemical stability was explained by difference in the abundance of chemical form obtained from the sequential extraction test.

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Keywords : Immobilization, Sintering, Radioactive waste, HyBRID, Decontamination

Evaluation of Dispersion Stability of Perfluorocarbon Based Decontamination Emulsion

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In a nuclear facility, the base metal can be radiologically contaminated during the operation. They must be decontaminated to reduce the radiation exposure to workers before decommissioning of the nuclear facility. In order to decontaminate the nuclear facility, it is possible to apply a perfluorocarbon (PFC) based emulsion consisted of surfactant and decontamination reagent. The PFC has high resistance for the radiation decomposition, and PFC based emulsion can be easily stabilized using the ultrasonication method. During decontamination process, a dispersion stability of the emulsion affects to the decontamination performance because the decontamination reagents dispersed in the emulsion contact contaminated surface. In this study, the dispersion stability the PFC based emulsion was evaluated following the composition of the emulsion and dispersion condition such as temperature, ultrasonication time. It was confirmed that the concentration of surfactant is highly related to the dispersion stability from the result of Turbiscan analysis using the multiple light scattering method. It was also verified that the droplet size of the decontamination reagent in the stable emulsion was smaller than that in the unstable emulsion. This phenomena can be explained by the relationship between the interfacial tension and droplet size. Finally, the recovering test of the PFC from the spent PFC-based decontamination emulsion was conducted using distillation method. The distillation test was performed using vacuum distillation unit, and the distillation temperature was 80°C. From the distillation test, about 95 % of PFC was recovered by distillation. From this result, it is considered that PFC-based decontamination emulsion reduces the volume of the secondary waste.

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Keywords: Decontamination, PFC, Emulsion, Dispersion stability, Recovery

Evaluation on Applicability of New Milling Machine Assisted Plasma Torch

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Various cutting technologies are being developed for dismantling nuclear power plants. these technologies are including mechanical and thermal methods. For example, mechanical cutting methods include sawing, drilling and milling. But, due to the strength of material, mechanical cutting methods have limits of cutting depth and tool life. Therefore, this milling machine assisted plasma torch was developed to improve the limits. And this machine has the principle of softening effect caused by the high temperature.

In this work, this developed device was evaluated in view of the cutting depth and tool life in cutting process. For this process, a plasma torch was attached to the front of the endmill processing path to heat the Inconel 600.

As results, compare to conventional milling, when the plasma torch power is 6.4 kW, the cutting depth was increased by 4 mm at condition (feed rate is $100 \text{ mm} \cdot \text{min}^{-1}$, tool diameter is 10 mm, rotating speed is 1,000 rpm). And cutting length increase 2 times from 300 mm to 600 mm at 16 mm of tool diameter.

Keywords: Plasma torch, Milling machine assisted plasma torch, Inconel 600, Cutting technology

Study on Applicability of Plasma Assisted Machining for Hard-Machining Materials

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Many countries are developing various mechanical cutting technologies to dismantle nuclear facility. However, most of mechanical cutting technologies have a problem like the degradation of tool life due to the Hard-Machining materials. To solve this problem, lab-scale test was performed with a Plasma Assisted Machining (PAM) technology and 25 mm of thickness Inconel 600 plate.

Commonly, the strength of metals decreases by exposure at high temperature. And, previous study reported that strength of Inconel 600 is degraded above 500°C. This softening effect was applied to Inconel 600 cutting test. The optimal conditions such as the plasma torch power and the feed rate were determined by this study.

As a result, the surface temperature of Inconel 600 was reached up to 500°C under the conditions which is 8.4 kW of plasma torch power and 150–250 mm·min⁻¹ of feed rate. And it was confirmed that the tool life was improved under the conditions. In order to apply PAM for various Hard-Machining materials, it is necessary to investigate the softening temperature of Hard-Machining materials, the plasma torch power and feed rate.

Keywords: Plasma torch, Plasma assisted machining (PAM), Inconel 600, Hard-Machining material

A Study on Cutting Technology for Dismantling Metal Structures of Nuclear Power Plant

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Currently, dismantling technology for decommissioning nuclear power plants is being developed around the world. This study describes the cutting technology and one of the technologies being considered for the RV/RVI cutting of Kori Unit 1.

The dismantling technology for nuclear power plants include mechanical and thermal methods. Mechanical cutting methods include milling, drill saw, and wire cutting. The advantages of the mechanical method are less generating aerosol and less performance degradation in water. However, the cutting speed is slow and the reaction force is large. Thermal cutting methods use heat sources such as plasma arcs, oxygen, and lasers. The advantages of thermal method are fast cutting speed, low reaction force and thick material cutting. On the other hand, they have problems with fume and melt. Among them, the cutability of the oxygen cutting method is better in carbon steel than in stainless steel.

In order to cut the RV/RVI of the Kori Unit 1, the applicability of fine plasma, arc saw, and band/wheel saw is being reviewed. For RV cutting, the applicability of arc saw and oxy-propane is being considered. Because RV is mostly made of carbon steel. However, since the flange is clad with stainless steel, the use of mechanical methods such as wire saws should be considered. In the case of RVI, since it has a complicated shape and is made of stainless steel, it seems necessary to review various cutting methods. In addition, it will be necessary to minimize radiation exposure of workers by cutting underwater cutting.

Keywords: Decommissioning nuclear power plant, Dismantling technology, Mechanical cutting, Thermal cutting

Waste Resin Treatment Generated From HWR Using Microwave Method

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Inorganic and organic ion exchange materials were generally applied to liquid processes in nuclear reactor. In the case of heavy-water reactor (HWR), zeolite, active carbon, anion resin, and cation resin were used to treat liquid processes such as reactor primary coolant cleanup and liquid radioactive waste management system. Then, used ion exchangers were stored at storage tanks. Various kinds of nuclides were adsorbed in ion exchange materials. Especially, C-14, long half-life nuclide, was highly concentrated in anion resin, and waste resin was treated as intermediated level radioactive waste (ILW). Thermal and non-thermal methods such as pyrolysis, incineration, catalytic extraction, acid digestion, and wet oxidation have been studied for treating spent resin. However, destructive methods are not suitable due to massive off gas waste containing radioactive species. To solve this problem, various kinds of processes were developed such as acid stripping, PLO process, activity stripping, thermal treatment, and etc.

In this study, microwave method is suggested to treat HWR waste resin. C-14 nuclide was selectively removed from waste resin without decomposition of main structure in waste resin. Radioactive waste resin generated from Wolsung HWR unit 1 and unit 2 was treated using microwave method and 95% of C-14 was successfully removed from the radioactive waste resin.

Keywords: C-14, Microwave, HWR, ILW, Waste resin

Hexagonal Boron Nitride Nanosheets for Enhanced of Tritium Separation in Water Electrolysis

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Water electrolysis is a representative technology for tritium enrichment in water. Proton exchange membrane (PEM) water electrolysis has received great attention to replace traditional alkaline water electrolysis which generates concentrated tritiated water containing a large amount of salts. Nafion has been widely used as a polymeric electrolyte for the PEM electrolyzer. However, its low gas barrier property causes explosion, corrosion or degradation of electrolyzer. Furthermore, the traditional polymeric electrolytes have negligible differences in conductivity between hydrogen isotopes. To enhance the tritium separation by water electrolysis, we designed a composite membrane (Nafion/hexagonal boron nitride (hBN)). The monolayer hBN has a high proton conductivity and gas barrier property, and the hBN can enhance conductivity differences between hydrogen isotopes. We prepared Nafion/hBN composite membranes, and water electrolysis performances and proton/deuterium separation behaviors were investigated.

Keywords: Tritium, Hydrogen isotope, Hexagonal boron nitride, 2D material, Water Electrolysis

A Study on Solvent Selection for Ligand Liquefaction in Decontamination Process Using Supercritical Carbon Dioxide

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As the number of nuclear power plants whose design life has expired worldwide increases, the attempts are continuing to complete the project of nuclear back-end cycle, the last task of the nuclear industry. Decontamination is essential in the process of dismantling nuclear facilities and restoration sites to remove all or some of the regulatory controls from an authorized facility. Among radioactive wastes, particularly contaminated soil is characterized by difficult physical decontamination because radionuclides are adsorbed between soil particles, that is, pores. Therefore, chemical decontamination is mainly used, which has the disadvantage of generating a lot of secondary waste.

In order to overcome these disadvantages, an eco-friendly soil decontamination process is being developed that can drastically reduce the amount of secondary waste generated by using supercritical carbon dioxide. Supercritical carbon dioxide can easily control its physical properties and has both liquid and gas properties. However, since supercritical carbon dioxide is non-polar, additives are needed to extract polar metal ions, which are the goal of decontamination. Therefore, ligand with both CO₂-philic and metal binding regions was selected.

In previous studies, the decontamination efficiency of soil was evaluated by reacting contaminated soil with solid ligand and co-ligand at once. When solid ligands were used, the decontamination efficiency was lower than expected, which was expected because chemical substances were somewhat difficult to exchange in the closed process.

In this study, in order to increase the efficiency of the decontamination process, the need for a process of liquefying ligand and continuously flowing it has been raised. Therefore, a co-solvent that dissolves well at the same time in SCCO₂, ligand, and co-ligand was selected. In the selection process, a total of eight substances were selected by dividing into six polar substances and two non-polar substances through various criteria such as economic feasibility, eco-friendliness, and harmlessness. Thereafter, ethanol was finally selected through solubility evaluation for SCCO₂ and additives. It is expected that a more effective decontamination process can be constructed when the additive is liquefied using a solvent selected from the results of this study.

Acknowledgements

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Keywords: Decommissioning, Soil decontamination, Supercritical CO₂, Eco-friendly, Chelating ligand, Solubility evaluation

Development of Integrated Algorithm Based on Deep Learning and Stereo Vision Technology for Real Time Worker Tracking in Decommissioning Site

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It is important to ensure worker's safety from radiation hazard in decommissioning site. Real-time tracking of worker's location is one of the factors necessary to detect radiation hazard in advance.

In this study, the integrated algorithm for worker tracking has been developed to ensure the safety of workers. There are three essential techniques needed to track worker's location, which are object detection, object tracking, and estimating location (stereo vision). Above all, object detection performance is most important factor in this study because the performance of tracking and estimating location is depended on worker detection level. YOLO (You Only Look Once version 5) model capable of real-time object detection was applied for worker detection. Among the various YOLO models, a model specialized for person detection was considered to maximize performance. This model showed good performance for distinguishing and detecting workers in various occlusion situations that are difficult to detect correctly. Deep SORT (Simple Online and Realtime Tracking) algorithm which uses deep learning technique has been considered for object tracking. Deep SORT is an algorithm that supplements the existing SORT method by utilizing the appearance information based on deep learning. It showed good tracking performance in the various occlusion situations. The last step is to estimate worker's location (x-y-z coordinates). The stereo vision technique has been considered to estimate location. It predicts xyz location using two images obtained from stereo camera like human eyes.

Two images are obtained from stereo camera and these images are rectified based on camera calibration information in the integrated algorithm. And then workers are detected from the two rectified images and the Deep SORT tracks workers based on worker's position and appearance between previous frames and current frames. Two points of workers having same ID in two rectified images give xzy information by calculating depth estimation of stereo vision.

The integrated algorithm developed in this study showed sufficient possibility to track workers in real time. It also showed fast speed to enable real-time application, showing about 0.08 sec per two frames to detect workers on a laptop with high-performance GPU (RTX 3080 laptop version). Therefore, it is expected that this algorithm can be sufficiently used to track workers in real decommissioning site by performing additional parameter optimization.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) Grant funded by the Ministry of Science and ICT (NRF-2020M2C9A1068162 and 2021M2E3A3040093).

Keywords: Object detection, Object tracking, Stereo vision, Computer vision, Decommissioning site, YOLO, Deep sort

Preliminary Evaluation of Drum Shielding Thickness and Number of Drums Required for Disposal of Spent Filter Stored in Kori NPP

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The spent filters stored in Kori Unit 1 are planned that compressed and disposed for volume reduction. However, shielding reinforcement is required to package high-dose spent filters in a 200 L drum. So, in this study suggests a shielding thickness that can satisfy the surface dose criteria of $10 \text{ mSv}\cdot\text{h}^{-1}$ when packaging several compressed spent filters into 200 L drums, and the number of drums required for the compressed spent filter packaging was calculated.

In this study, representative gamma-emitting nuclides in spent filter are assumed that Co-60 and Cs-137, and dose reduction due to half-life is not considered, because the date of occurrence and nuclide information of the stored spent filter are not accurate. The shielding material is assumed to be concrete, and the thickness of the shielding is assumed to 18 cm considering the diameter of the spent filter and compression mold.

Considering the height of the compressed spent filter and the internal height of the shielding drum, assuming the placement of the compressed spent filter in the drum in the vertical direction only, the maximum number of packaging of the compressed spent filter is 3. When applying a 18 cm thick concrete shield, the maximum dose of the spent filter can packaged in the drum is $125 \text{ mSv}\cdot\text{h}^{-1}$, so when packaging 3 spent filters of the same dose, the dose of a spent filter shall not exceed $41 \text{ mSv}\cdot\text{h}^{-1}$ and not exceed $62 \text{ mSv}\cdot\text{h}^{-1}$ when packing 2 spent filters.

Therefore, the dose ranges of spent filters that can be packaged in a drum are classified into three groups: $0\text{--}41 \text{ mSv}\cdot\text{h}^{-1}$, $41\text{--}62 \text{ mSv}\cdot\text{h}^{-1}$, and $62\text{--}125 \text{ mSv}\cdot\text{h}^{-1}$ based on $41 \text{ mSv}\cdot\text{h}^{-1}$, $62 \text{ mSv}\cdot\text{h}^{-1}$, and $125 \text{ mSv}\cdot\text{h}^{-1}$. When 227 spent filters stored in the filter room are classified according to the above dose group, 207, 3 and 4 spent filters are distributed in each group, and the number of shielding drums required to pack the appropriate number of spent filters in each dose group is 75. Meanwhile, 8 spent filters exceeding $125 \text{ mSv}\cdot\text{h}^{-1}$ and 5 spent filters that has without dose information are excluded from compression and packaging until the treatment and disposal method are prepared.

In the future, we will segmentation of waste filter dose groups through the consideration of dose reduction and horizontal placement of compressed spent filters, and derive the minimum number of drums required for compressed spent filter packaging.

Keywords: Spent filters, Waste acceptance criteria, Drum shielding thickness, Dose group

Solidification Properties of Magnesium Phosphate Cement for Cesium, Strontium, and Cobalt

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Magnesium potassium phosphate cements (MKPCs) are prepared by the acid-base reaction of dead burned magnesia (MgO) and monopotassium phosphate (KH_2PO_4). Low-pH cementitious materials such as MKPCs are currently of interest for the geological disposal of nuclear waste. MKPCs have advantages such as high early strength, high bonding strength, small drying shrinkage, low permeability, and high sulfate resistance. According to the results of previous studies, it is known that cesium, strontium, and cobalt are immobilized in the form of $\text{MgCs}_x\text{K}_{1-x}\text{PO}_4 \cdot 6\text{H}_2\text{O}$, $\text{Mg}_x\text{Sr}_{1-x}\text{KPO}_4 \cdot 6\text{H}_2\text{O}$, and $\text{Co}_3(\text{PO}_4)_2$, respectively, in MKPCs. However, these results were predicted based on thermodynamic data, not directly observed precipitates to clearly show the evidence. Therefore, in this study, we directly analyzed the immobilized forms of Cs, Sr, and Co, respectively. CsNO_3 , $\text{Sr}(\text{NO}_3)_2$, and $\text{Co}(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ powders (0.3 mol each) were mixed individually in each of the MKPC suspensions. The suspensions in which KH_2PO_4 was dissolved were pH 4.3 and the dissolution of MgO decreased the H^+ concentration, raising the pH close to 11. The hydration products according to pH evolution in the MKPC suspensions were analyzed, and the change in the concentration of ions in the aqueous solution was also measured. An aqueous solution was obtained using a syringe filter (0.45 μm) to analyze the ion concentrations in the solution of the suspension. The collected solutions were diluted with nitric acid and analyzed using inductively coupled plasma mass spectrometry. To characterize the solid phases, the suspensions were obtained with a pipette at specific times and filtered under a vacuum in a Buchner funnel. Because the amounts of hydration products including Cs, Sr, and Co were small, it was not observed by XRD and TGA analysis, but their components could be analyzed by SEM-EDS. The final precipitate forms of Cs, Sr, and Co in the MKPC matrix are $\text{MgCsPO}_4 \cdot 6\text{H}_2\text{O}$, SrHPO_4 , and $\text{Co}_3(\text{PO}_4)_2 \cdot 8\text{H}_2\text{O}$, respectively.

Keywords: Magnesium potassium phosphate cement (MKPC), Solidification, Immobilization, Cement hydration

A Comparative Analysis of Derived Concentration Guideline Levels for the Site Release of Nuclear Power Plant After Decommissioning

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The Kori Unit 1 and Wolsong Units 1, commercial reactors in South Korea, were permanently shut down due to the expiration of their design lifetime. Therefore, nuclear power plants that have been permanently shut down must be dismantled, and the site must be finally released after removing the remaining radionuclides. Domestic regulatory standards for site remediation should not exceed 0.1 mSv per year based on effective dose. In addition, it is necessary to calculate the preliminary Derived Concentration Guideline Levels (DCGL) to prove that the conditions are met. Therefore, in this study, the input factor considering the geological characteristics of the site of Kori Unit 1 was investigated, and the preliminary Derived Concentration Guideline Levels were calculated and compared with the results of previous studies.

As a result of comparative analysis, ^{60}Co , ^{134}Cs , and ^{137}Cs , which are gamma-ray emitting radionuclides, had similar values to DCGL of previous studies A and B. However, ^{63}Ni , a beta-ray-emitting nuclide, was $5.94 \times 10^4 \text{ Bq} \cdot \text{g}^{-1}$ in this study and $8.47 \times 10^1 \text{ Bq} \cdot \text{g}^{-1}$ in previous study B, resulting in a difference of about 700 times. In addition, in the case of ^{90}Sr , this study and previous study A were derived similarly, but this study was $5.34 \times 10^1 \text{ Bq} \cdot \text{g}^{-1}$ and previous study B was $1.18 \times 10^{-1} \text{ Bq} \cdot \text{g}^{-1}$, resulting in a difference of about 450 times. This difference is judged to be because, unlike this study using only the industrial worker scenario, in the case of previous study B, the resident farmer scenario was mixed and used, which considers the internal exposure caused by ingestion of food produced in the contaminated area.

In this study, it was confirmed that DCGL according to the change of geological factors of the site did not have a significant effect on gamma-ray-emitting nuclides. However, it was confirmed that considering the intake of food affects the DCGL of beta-ray-emitting nuclides. Therefore, there is a need to conduct future studies applying intake input factors that meet domestic conditions.

Keywords: Site release, DCGL, Decommissioning, RESRAD-ONSITE

Co-SPND Initial Sensitivity Calculation and Activation Analysis With MC Simulation

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Self-Powered Neutron Detector (SPND) is one of devices for in-core fluxes detecting without external electricity source. SPND consisted with emitter, insulator and collector. When neutrons reacted with emitter material, it generates electrons and these electrons cross insulator area to make electric signal in collector area. For calculating sensitivity of SPND with Monte-Carlo code such as MCNP, many physical components must be considered. Cobalt shows that prompt signal and relatively low signal comparing with other delayed signal SPNDs. Initial sensitivity was calculated as 4.28×10^{-22} A/nv-cm for one electron.

Due to Cobalt's complex decay chain and maintaining high efficiency of SPND, it is necessary to analysis the effect of activation of emitter. Therefore, the DPA (Displacements Per Atom) assessment and activation analysis of the detector components have been evaluated with MCNP 6.2 and ORIGEN-S. With these activation analysis results, that is expected to be used to determine the shielding thickness of the storage system.

Keywords: SPND, MCNP, DPA, ORIGEN

Treatment of Wastewater From the Foam Decontamination Process of Contaminated Metal Components

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Radioactively contaminated metal components from a nuclear power plant must be decontaminated to reduce the risk of radiation exposure to workers, which can be cleaned using a foam decontamination used to reduce the amount of wastewater significantly. Metal components with a fixed radioactive contamination can be effectively decontaminated using a foam consist of 0.5wt% nonionic surfactant, 0.5 M H₂SO₄, and 0.2 M Ce(SO₄)₂. However, strongly acidic wastewater is generated from the decontamination method, which contains a high concentration of the nonionic surfactant and ionic materials with radioactive nuclides. This wastewater must be treated as a stable form. In this study, an integrated process of precipitation and low pressure distillation was evaluated for the treatment of wastewater. It was confirmed that the surfactant and ionic materials were effectively removed from the wastewater through the integrated process.

Keywords: Radioactively contaminated metal, Foam decontamination wastewater, Surfactant, Ionic material, Precipitation, Low pressure distillation

Immobilization of C-14 Waste by Cement Wasteform

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Radioactive carbon, C-14, can be generated by the neutron capture reaction of O-17 during the nuclear power plant operation. Since C-14 is classified as an intermediate level waste radionuclide, it is required that an effective separation process for C-14. C-14 is mainly absorbed on activated carbon in the air cleanup system. Therefore, the main generation source of C-14 during the nuclear power plant decommissioning is spent activated carbon. KAERI has been developing the treatment of spent activated carbon. In this process, C-14 can be desorbed as a gaseous oxide form from the spent activated carbon at high-temperature vacuum conditions. This radioactive carbon dioxide can be captured into alkaline earth metal incorporated glass and can be transformed into carbonate form. However, the carbonate (e.g. CaCO_3 and SrCO_3) is dispersive. When the radioactive carbonates are disposed into a geological repository, they should be immobilized to remove future uncertainty. This study examined the stabilization/immobilization of the radioactive carbonates by the cement hydration process. Cement wasteform incorporated with calcium carbonate and strontium carbonate was produced under various waste loading (e.g. 20wt%, 40wt%, and 60wt% of CaCO_3 and SrCO_3 , respectively). Then we evaluated mechanical and chemical durability by measuring compressive strength and leachability according to standard test methods specified in the waste acceptance criteria of the Gyeongju low and intermediate level waste repository (WAC-SIL-2022-1). Also, microstructure and thermal characteristics were investigated by SEM-EDS and TGA analysis.

Keywords: C-14, Carbon waste, Cement hydration, Carbonate, Wasteform

Virtual Manipulator System Using a Haptic Device for Remote Dismantling of Nuclear Reactor

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Since nuclear power plant (NPP) dismantling carries the possibility of radiation exposure from a hazardous environment, it's important to minimize that by using a remote manipulator et al. However, due to complexity of nuclear facilities, it's necessary for operators to increase their proficiency by operating in advance in a virtual environment. In this research, we propose a virtual manipulator system using a haptic device for NPP's reactor vessel internals (RVI) dismantling which can realistically manipulate.

Acknowledgements

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Keywords: Nuclear power plant, Virtual manipulator, Haptic device, Reactor vessel internals

Evaluation of Chemical Precipitation for ^{137}Cs and ^{90}Sr Separation From Complexed Acidic Wastewater

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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, the volume of radioactive concrete waste could be reduced effectively. We were studied to separate the radioactive materials from the concrete by using the thermomechanical and chemical treatment processes, sequentially. From the study, separated aggregate could be treated to achieve the clearance level. However, these processes generate a large volume of secondary acidic radioactive wastewater, which might be a critical problem to reduce the volume of radioactive concrete waste.

In this research, separating the ^{137}Cs and ^{90}Sr from dissolved concrete wastewater to below the discharge criteria by precipitation method, it would be released to the environment under industrial waste guidelines. The experiments were conducted to using a simulated radioactive wastewater, formed by the dissolution of concrete within HCl, which was spiking the ^{137}Cs and ^{90}Sr , respectively. In addition, we applied the chemical precipitation methods with wastewater, using ferrocyanide for ^{137}Cs and BaSO_4 coprecipitation for ^{90}Sr . As a result, targeted radionuclides could be removed to the discharge level (^{137}Cs : $0.05 \text{ Bq}\cdot\text{ml}^{-1}$, ^{90}Sr : $0.02 \text{ Bq}\cdot\text{ml}^{-1}$) by precipitation method. Therefore, it could reduce the secondary wastewater effectively by precipitation method and enhance the additional volume reduction for radioactive concrete waste.

Keywords: Radioactive concrete waste, Volume reduction, Chemical precipitation, Discharge criteria

Effect of Additives for the Stabilization of Radioactive Soil Waste by Glass Composite Materials

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As the design life of nuclear power plants are coming to the end, starting with Kori unit 1, nuclear power related organizations have been actively conducted research on the treatment of nuclear power plant decommissioning waste.

In this study, among various types of radioactive waste, stabilization and volume reduction experiments were conducted on radioactive contaminated soil waste. Korea has no experience in decommissioning nuclear power plants, but a large amount of radioactively contaminated soil waste was generated during the decommissioning of the KAERI research reactor (TRIGA Mark- II) and the uranium conversion facility. This case shows the possibility of generating radioactive soil waste from nuclear power plants and nuclear-related facilities sites.

Soil waste should be solidified, because its fluidity and dispersibility wastes specified in the notification of the Korea Nuclear Safety and Security Commission. In addition, the solidified waste forms should have sufficient mechanical strength and water resistance.

Numerous minerals in the soil are components that can make glass and ceramics, for this reason, glass-ceramic sintered body can be made by appropriate heat and pressure. The sintering conditions of soil were optimized, in order to make better economical and more stable sintered body, some additives (such as additives for glass were mixed) with the soil and sintering experiments were conducted.

Uncontaminated natural soil was collected and used for the experiment after air drying. Moisture content, pH, bulk density, and organic content were measured to understand the basic properties of soil, and physicochemical properties of the soil were identified by XRD, XRF, TG, and SEM-EDS analysis. In order to understand the distribution by particle size of the soil, it was divided into Sand (0.05–2 mm) and Fines (< 0.05 mm). The green body was manufactured in the form of a cylinder with a diameter of 13mm and a height of about 10mm. Appropriate pressure (> 150 MPa) was applied to the soil to make a green body, and appropriate heat (> 800°C) was applied to the sintered body to make a sintered body. The sintering was conducted in a muffle furnace in air conditions.

The volume reduction and compressive strength of the sintered body for each condition were evaluated.

Keywords: Radioactive waste, Volume reduction, Additive, TGA, Compressive strength

A Method for Optimization of Radioactively Contaminated Soil Remediation Technology Using Digital Twin

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Canceled Poster

A Method for Developing 3D Based Nuclear Power Plant Decommissioning System

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Considering the characteristics of nuclear power plants in order to decommission nuclear power plants safely and economically, this thesis provides a methodology for optimizing the technology for developing decommissioning characteristic evaluation system using simulation technology for core facilities of the plants based on 3D that reflects various factors. The results of pollution assessment and radiation assessment for the Kori Unit 1 reactor building, auxiliary building, and each major device are displayed in 3D drawings and viewer, and the radiation dose rate and radiation assessment results are displayed separately for each major location.

Furthermore, this D/B development method which includes inserting result values of characteristic evaluation and the quantity of waste is one of the main technology to optimize the system which enables users to select decommissioning processes and predict the quantity of waste. (Refer to the presented 3D models of the containment building, D/B, tag search module, the scale calculation result of models after visualizing the result value of 3D based decommissioning characteristic evaluation)

The methodology for optimizing decommissioning characteristic evaluation result value DB development system using 3D models of the first major nuclear power plant allows the display of decommissioning characteristic values in virtual reality, the selection of decommissioning process, the establishment of the decommissioning procedure. Hence, this study is expected to provide reliable guidelines for managing a decommission business efficiently in the near future and can be used in the related field if needed.

Keywords: 3D Model, Nuclear power plant, Decommissioning, Radioactivity

6분과

방사선환경 및 안전(Oral)

Technical Review on the Scene Data Fusion

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Gamma Reality Inc. (GRI) provides real-time, mobile 3D radiation mapping, data fusion, and visualization technologies for applications ranging from nuclear power and decommissioning to emergency response. The GRI-LAMP is a compact, multi-sensor system weighing about 10 lbs (4.5 kg). LAMP is fully mobile, provides 360 degree imaging (only limited by physical access to objects/area), and streams the 3D map fused with radiation data in real-time to the control tablet for immediate results that can quickly inform the user of potential hazards in the area or direct the user to the specific location where efforts should be focused. GRI systems are also remotely deployable on robotic platforms and are used on unmanned aerial vehicles (UAS), unmanned ground vehicles (UGV), as well as on manned vehicles and in handheld configurations. This deployment flexibility coupled with real-time data maximizes dose reduction opportunities and further enables dynamic operational planning, which can help reduce the costs of managing and maintaining operational nuclear power plants, as well as decontaminating, or decommissioning nuclear facilities. Applications include, but are not limited to, conducting regular radiation surveys, hotspot localization, shielding verification, radioactive waste shipment surveys, contamination mapping, and dose measurement. GRI's solutions enable faster, safer, and more efficient radiation detection, mapping, and visualization of source terms and contamination. Commercially available LAMP versions include gamma-ray imaging, dual neutron and gamma mapping, and non-imaging gamma-ray mapping options.

Keywords: Scene data fusion, Gamma-ray imaging, Source localization, Radiation mapping

A Preliminary Study on the Estimation of Cs-137 Source Positions Using Double-Trained Machine Learning Model

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In this study, the positions of Cs-137 gamma ray source are estimated from the plastic scintillating fiber bundle sensor with length of 5 m, using machine learning data analysis. Seven strands of plastic scintillating fibers are bundled by black shrink tube and two photomultiplier tubes are used as a gamma ray sensing and light measuring devices, respectively. The dose rate of Cs-137 used in this study is $6 \mu\text{Sv}\cdot\text{h}^{-1}$. For the machine learning modeling, Keras framework in a Python environment is used. The algorithm chosen to construct machine learning model is regression with 15,000 number of nodes in each hidden layer.

The pulse-shaped signals measured by photomultiplier tubes are saved as discrete digits and each pulse data consists of 1,024 number of them. Measurements are conducted separately to create machine learning data used in training and test processes. Measurement times were different for obtaining training and test data which were 1 minute and 5 seconds, respectively. It is because sufficient number of data are needed in case of training data, while the measurement time of test data implies the actual measuring time. The machine learning model is designated to estimate the source positions using the information about time difference of the pulses which are created simultaneously by the interaction of gamma ray and plastic scintillating fiber sensor.

To evaluate whether the double-trained machine learning model shows enhancement in accuracy of source position estimation, the reference model is constructed using training data with one-time learning process. The double-trained machine learning model is designed to construct first model and create a second training data using the training error and predetermined coefficient. The second training data are used to construct a final model. Both reference model and double-trained models constructed with different coefficients are evaluated with test data. The evaluation result shows that the average values calculated for all measured position in each model are different from 7.21 to 1.44 cm. As a result, by constructing the double-trained machine learning model, the final accuracy shows 80% of improvement ratio.

Further study will be conducted to evaluate whether the double-trained machine learning model is applicable to other data obtained from measurement of gamma ray sources with different energy and set a methodology to find optimal coefficient.

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Keywords: Machine learning, Plastic scintillating fiber, Position estimation, Double-trained modeling

Qualitative and Quantitative Comparative Analysis of Radioactivity Distribution in Concrete Shielding Wall

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The amount of radioactive waste generated during decommissioning directly affects the disposal cost of waste. Most of the radioactive waste generated is a concrete waste. Therefore reducing the amount of concrete waste can ensure the economic feasibility of the decommissioning project. The activated concrete in a concrete waste can reduce waste only by physical cutting. Therefore it is most important to accurately identify and categorize radionuclides, radioactivity levels, and radioactivity distribution.

In the case of radioactive concrete, radiological characteristics are generally evaluated by laboratory analysis after sampling. However it is difficult to apply to all facilities (accelerator & NPP, etc.) because it is a destructive method. Therefore it is necessary to secure verified in-situ measurement technology that can be applied to operational monitoring or decommissioning plans. In this study, the applicability of cyclotron facilities was evaluated based on the evaluation algorithm derived from the Peak to Compton (PTC) method of in-situ measurement technology. And the reliability of the PTC method was verified through qualitative analysis and quantitative analysis. In the case of qualitative analysis, the analysis results of KAERI which has core technology are compared. To this end SAE-AN and KAERI conducted field application tests on the front concrete shielding wall of the cyclotron facility at the same time.

After removing the background spectrum from the measured spectrum the PTC method was applied to calculate the Q-value for the counting rate in the peak area per counting rate in the Compton continuum area was calculated. As a result the Q-values of SAE-AN and KAERI were 0.52 and 0.24 respectively, and the result of deriving activation distribution(β) by substituting this for the β -Q correlation equation was found that 14.78 and 12.94. As a result of evaluating the activation by the thickness of the shielding wall it was found that 89.1% (SAE-AN) and 91.9% (KAERI) of the total radioactivity were exist at a depth of 5 cm. And it was found that 97.7% and 99.05% of the total radioactivity exists at a depth of 10 cm. The relative error between SAE-AN and KAERI is 1.35%, indicating that the analysis results of the two institutions are highly consistent.

A core drill was performed on the concrete shielding wall in the cyclotron facility for the technical verification of the quantitative analysis method. A core sample (6 cm in diameter, 10 cm in depth) was cut to a depth of 2 cm and analyzed in the laboratory. The activation distribution(β) was calculated based on the radioactivity level of each depth sample, and it was found to be 16.99. The relative error between the quantitative analysis and the on-site measurement results was 14.95% confirming that the accuracy is relatively high.

Keywords: In-Situ, Radioactive concrete, Peak to compton, Depth, reliability

Development of Virtual Gamma-ray Energy Spectrum Database for KHNP Detectors by MCNP Simulation

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To obtain the gamma-ray energy spectrum of artificial radionuclides which is difficult to obtain practically, virtual gamma-ray energy spectrum simulator program was developed. It can be applied for the predetermined measurement condition for which the database was developed through computational simulation and actual measurement of background radiation. For gamma spectrometry training for KHNP HPGe detectors using this program, the database for KNPG HPGe detectors was developed. First, the geometry of the detector in the simulation was adjusted to resemble the real structure by comparing the actually measured net counts rate at the main gamma peak with the value simulated by MCNP6. The Certified Reference material (CRM) of ^{137}Cs and ^{60}Co were used for verification. The comparison was made with respect to the situation where CRM was attached to the top and side of the detection part of the considered detector. The geometry structures of detectors were simulated by reflecting the design drawing of the products, and the simulation was performed for several thicknesses of the Ge/Li dead layer in consideration of the change in the thickness over time. As the results, the simulation geometry was tuned so that the results for ^{137}Cs showed a difference within 10% for all detectors. At this time, in some detectors, the result for ^{60}Co shows a 10% higher error, which is estimated to be due to the random summing. It was not considered in tuning the simulation geometry, but it was found that improvements were needed to reflect the coincidence summing when construction the virtual spectrum in the future.

The determined simulation geometry was applied to generate theoretical gamma-ray energy spectra of representative artificial radionuclides. In order to create a virtual spectrum similar to the real one, the background spectrum was measured for each detector without a source, and the simulation results were calculated in the form of having the same energy channel as the background spectrum. The background spectrum and theoretical spectra of artificial radionuclides for each detector were databased so that virtual spectra could be generated under desired conditions. The virtual spectrum was generated by adding a background spectrum and a spectrum obtained by multiplying the spectrum of the desired nuclide by the concentration of the nuclide. The validity of generated virtual spectra was verified using the pre-developed gamma spectrometry program. As a results of gamma spectrometry of virtual spectra, the virtual spectra was verified by showing a difference within 20% from the radioactivity value input when generating the virtual spectra.

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Keywords: In-situ gamma spectrometry, MCNP, HPGe detector, Virtual spectrum

Potential of Tin Slag-containing Aggregates as Road Pavement Material

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Tin slag is a byproduct obtained from the tin smelting industry and contained naturally occurring radioactive material (NORM); therefore, it has to be managed accordingly. This study focuses on recycling the waste in exchange for natural aggregates for road pavement due to the potential features as construction materials. The main objective of this study is to analyze the use of tin slag by diluting its radioactivity level and as the replacement of natural aggregates while focusing on identifying the mechanical properties of the mixture. Tin slag was used as coarse aggregate in the range of 0–85% while the percentage of recycle glass was maintained at 15% and granite rocks in range of 0–100%. In this research, the concentration activity of NORM in tin slag have been measured using a gamma ray spectrometer. Few laboratory tests for the final product are carried out such as Los Angeles abrasion value (LAAB), aggregate crushing value (ACV), and aggregate impact value (AIV). This study was also conducted to analyze the leachability of As, Cd, Ba, Cr, Pb, Se and Ag from the different composition. From the measurement result, the average concentration of ²²⁶Ra, ²³²Th and ⁴⁰K are 318.21 Bq·kg⁻¹, 602.07 Bq·kg⁻¹ and 89.84 Bq·kg⁻¹, respectively. The outdoor dose rates were found to be lower than 1.5 mSv·yr⁻¹ in sample A1, A2 and A3 which is the recommended limit for construction materials. The sample toxicity was assessed using the toxicity characteristic leaching procedure (TCLP) and the concentration of the elements studied was analysed using ICP-MS. The result from the analysis indicated that the concentrations of the heavy metal elements were between 0.001–26.94 mg·kg⁻¹, which is lower than the limit for each element. As a conclusion, addition of tin slag between 5 to 25% in exchange of granite rocks as road pavement have showed potential evidence in the test for construction material. Besides, it has low leachability to the environment while diluting the radioactivity level.

Keywords: NORM, Tin slag, Leachability, Dose rate, Radioactivity, Heavy metals

Criteria for Safe Handling and Disposal of NORM Residue

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In Malaysia, there are several industries processing mineral ores generate residues containing naturally occurring radioactive material (NORM) with activity concentrations above the control limits established by the Malaysian Atomic Energy Licensing Board (AELB). These industries use mineral ores or concentrated ores as their feed materials to produce or extract valuable sand minerals or rare earth compounds for use in another industries. The control limits for activity concentrations of Uranium-238 (U-238) and Thorium-232 (Th-232) and their decay series is 1.0 Becquerel per gram ($\text{Bq}\cdot\text{g}^{-1}$) while activity concentration of Potassium 40 (K-40) is 10.0 $\text{Bq}\cdot\text{g}^{-1}$. The management of residue containing NORM radioactivity above the control limits must be done in accordance with current rules and regulations including proper handling, storage, transportation and/or disposal. Where possible, appropriate mixture process with other non-radiological material would reduce the activity concentrations to below the control limits. Depending on specific characteristics of residue, appropriate approach to reuse or recycle should be encouraged as part of special waste management. For this case, an exemption to release it from radiological controls can be applied but require scrutiny review and approval process by AELB. In addition, the health and safety aspects and environmental issues should be assessed which to be done in accordance with the relevant rules and regulations. As a last resort, a disposal of residue containing NORM radioactivity shall be done at the landfill disposal facility approved by AELB and other relevant Authorities.

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Atomic Energy Licensing Board, Malaysian Nuclear Professional Chapter.

Keywords: Naturally occurring radioactive material, U-238, Th-232, Control limit

A Ship-mounted Remote Real-time Gamma-ray Monitoring System of Seawater

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Recently, an international issue due to the discharge of contaminated water from the Fukushima has been highlighted. Since the Fukushima nuclear power plant accident in Japan, marine environmental radioactivity survey has been strengthened with increased sampling frequency and range for seawater in territorial waters. And a stationary underwater radiation monitoring system including floating equipment-based system such as oceanographic buoys, tidal stations have been deployed on-site to detect abnormal radiological events. However, stationary monitoring systems may be insufficient for the early detection of abnormal radioactivity over a wide area, since it is a passive way of waiting for radioactive materials to spread in the ocean. So, our team developed a ship-mounted seawater gamma-ray monitoring system that can be operated remotely and in real time. In this study, it includes a detailed description of the design, installation, monitoring method, and operation of the system.

Keywords: Ship-mounted, Gamma-ray monitoring system, Seawater, Contaminated water, Fukushima

Migration Behavior of Cesium in the Subsurface Environment After Wildfires

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Forest fires produce various particulate organic matters (POMs) derived from the incomplete combustion process of biomass. The POMs deposited in soil and sediments can affect the physicochemical properties of the subsurface environments. This study investigated the sorption and transport behavior of cesium (Cs) in soil-groundwater environment after wildfire.

Soil samples were collected at two locations (GS1 & GS2) in Gangwon Province, Korea, at different depths (~5, ~20, and ~40 cm). The sampling site, where a large-scale forest fire occurred in 2017, was damaged almost 252 ha of forest. The soil characteristics were determined by X-ray diffraction (XRD), X-ray fluorescence (XRF), scanning electron microscopy (SEM), total organic carbon (TOC) analysis and organic petrography, and batch and fixed-bed column experiments were performed to evaluate the Cs uptake and retardation.

The XRD patterns of the soils indicated that the mineral compositions of soils were quartz, feldspars (e.g., orthoclase & albite) with minor muscovite/illite. Quartz and feldspars were abundant in all studied soils, and GS2 sample contained higher feldspars and phyllosilicate minerals than the GS1. The TOC contents were high (7–8wt%) in the topsoils, decreasing with depth. The SEM and organic petrographic analyses showed that various organic carbon particles such as textinite, ulminite, fusinite (charcoal) and char existed. Presence of charcoal and char is the evidence of wildfires, even though their amount was few. Batch sorption experiments revealed that the K_d value decreased non-linearly as the Cs concentrations increased, and the sorption isotherms were fitted well with the Freundlich model. The K_d values of each soil were much greater in topsoils compared to subsoils at all experimental Cs concentrations. In particular, the GS1 topsoil had higher sorption capacity for cesium than GS2 subsoils, although it had low phyllosilicate mineral contents with relatively rich organic matter. The breakthrough curve of column experiments with high concentration ($C_0 \approx 1 \times 10^5 \mu\text{g} \cdot \text{L}^{-1}$) also exhibited remarkable Cs retardation phenomena in topsoils. Their retardation factors ($R_{f,Cs}$) were max. 4 times greater than those of subsoils, showing $R_{f,Cs} \approx 43$ to 45 for topsoils. At low concentration ($C_0 \approx 1 \times 10^4 \mu\text{g} \cdot \text{L}^{-1}$), the $R_{f,Cs}$ of topsoils (≈ 284 to 374) was slightly greater than that of subsoils (≈ 270 to 271). These results imply that POMs caused by wildfires can play important role on the Cs sorption and transport in the subsurface environments.

Keywords: Wildfire, Cesium, Transport, Subsurface, Particulate organic matter

6분과

방사선환경 및 안전(Poster)

A New Approach to Pre-processing for Gross Alpha Measurements

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For the final disposal of radioactive waste generated during the operation of nuclear power plants, concentrations of 14 radionuclides including gross alpha have to be determined to meet nuclear regulatory requirements.

In order to determine the gross alpha radioactivity in radioactive waste, the sample must be preprocessed into a solution which is usually a strong acid.

When this solution is used to prepare the gross alpha measurement sample, it produces a lot of salt, which makes an accurate measurement difficult. Also it causes corrosion of a planchet, which causes problems in the disposal of waste in the future. For these reasons, an acid treatment of the solution was added to the existing preprocess procedure, which is also expected to improve the measurement error.

Although the gross alpha measurement is known to be easy to perform and able to give rapid results, it cannot be used for quantitative analysis. This is because the energies emitted by the individual alpha nuclides are assumed to be produced from a single alpha emitted by the individual alpha nuclides are assumed to be produced from a single alpha emitter used as the standard calibration source. Also, due to self-absorption of alpha particles a counting rate depends on the thickness (or weight) of the residues on the planchet.

In this study, we compared gross alpha radioactivity with and without an acid treatment to prepare gross alpha measurement samples. The weights of the treated samples increased by at most 5% after about 12 hours of evaporation to dryness, and then saturated or slightly decreased, while the weights of the untreated samples increased up to 20% over time. In addition, the radioactivities of the untreated samples were about two times those of the treated samples. This is considered to be due to differences both in the geometric shapes of the samples and the weights of their residues which resulted from whether acid treatment was applied or not. The results of this study showed that an acid treatment was beneficial in reducing both production corrosion and salts which could result in more reliable and constant measurements of gross alpha activity. The results showed that acid treatment was beneficial in reducing corrosion and measurement errors.

Keywords: Radioactive, Gross alpha, Radioactive waste

Determination of ^{210}Po in Bottled Drinking Water Using Alpha Spectrometry

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^{210}Po is a naturally occurring radionuclide of ^{238}U decay series with a half-life of 138.4 days. ^{210}Po is decay products of ^{222}Rn , which escapes into the atmosphere and present in all environments with aerosol particles. Also, ^{210}Po has high radiotoxicity and emits a high alpha energy of 5.305 MeV, and it decays to finally become a stable isotope, ^{206}Pb . Therefore, ^{210}Po entering the body by continuously ingestion or inhalation is likely to cause severe damage to the bone marrow, kidney and spleen and other sites in the body. Accordingly, the World Health Organization (WHO) recommends that screening level of gross alpha for drinking water not exceed $0.5 \text{ Bq}\cdot\text{L}^{-1}$.

Alpha spectrometry has been mainly used for analysis of ^{210}Po , and for the accurate measurement of alpha particle with short range, it is essential to prepare suitable source for alpha detection. The ^{210}Po alpha source is made by a spontaneous deposition method in which polonium is adsorbed thin and flat onto a metal disc, such as silver, nickel and copper. There are various pretreatment methods to separate and concentrate polonium from water samples prior to spontaneous deposition, including $\text{Fe}(\text{OH})_3$ or MnO_2 co-precipitation and evaporation. However, in the case of co-precipitation, sample contamination or loss of polonium may occur through the experimental processes, and evaporation lead to not only time-consuming process but also may cause loss of polonium due to the low boiling point of polonium. Therefore, in order to compensate for these problems, an efficient polonium analysis method that directly collects polonium from the original sample without a pretreatment is required.

In this study, ^{210}Po in bottled drinking water sold in Korea was analyzed using alpha spectrometry. A high purity silver disc (99.99%) was inserted into a newly designed polonium deposition kit to quickly and conveniently collect polonium from a water sample. The polonium alpha detecting source was made effectively only by the spontaneous deposition method without a complicated pretreatment. The source was measured using a PIPS detector, and the radioactivity concentration of ^{210}Po was calculated using ^{209}Po as a yield tracer.

Keywords: ^{210}Po , Bottled drinking water, Spontaneous deposition, Alpha spectrometry

Liquid Light Guide Based Radiation Sensor System for Remote Gamma-ray Spectroscopy

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Gamma-ray spectroscopy, which is an appropriate method to identify and quantify radionuclides, is widely utilized in radiological leakage monitoring of nuclear facilities, assay of radioactive wastes, and decontamination evaluation of post-processing such as decommissioning and remediation. For example, in the post-processing, it is conducted to verify the radioactivity level of the site before and after the work and decide to recycle or dispose the generated waste.

For an accurate evaluation of gamma-ray emitting radionuclides, the measurement should be carried out near the region of interest on site, or a sample analysis should be performed in the laboratory. However, the region is inaccessible due to the safety-critical nature of nuclear facilities, and excessive radiation exposure to workers could be caused. In addition, in the case of subjects that may be contaminated inside such as pipe structures generated during decommissioning, surveying is usually done over the outside of them only, so the effectiveness of the result is limited. Thus, there is a need to develop a radiation measurement system that can be available in narrow space and can sense remotely with excellent performance.

A liquid light guide (LLG), unlike typical optical fiber, is a light guide which has a liquid core. It has superior light transmissivity than any optical fiber and can be manufactured with a larger diameter. Additionally, it can deliver light with much greater intensity with very low attenuation along the length because there is no packing fraction and it has very high radiation resistant characteristics. Especially, thanks to the good transmissivity in UV-VIS wavelength, the LLG can well transmit the scintillation light signals from scintillators that have relatively short emission wavelengths, such as LaBr₃:Ce and CeBr₃.

In this study, we developed a radiation sensor system based on a LLG for remote gamma-ray spectroscopy. We fabricated a radiation sensor with LaBr₃:Ce scintillator and LLG, and acquired energy spectra of Cs-137 and Co-60 remotely. Furthermore, the results of gamma-ray spectroscopy using different lengths of LLG were compared with those obtained without LLG. Energy resolutions were estimated as 7.67%, 4.90%, and 4.81% at 662, 1,173, and 1,332 keV, respectively for 1 m long LLG, which shows similar values of a general NaI(Tl) scintillator. With 3 m long LLG, the energy resolutions were 7.92%, 5.48%, and 5.07% for 662, 1,173, and 1,332 keV gamma-rays, respectively.

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Keywords: Remote gamma-ray spectroscopy, Liquid light guide, LaBr₃(Ce), Cs-137, Co-60

Improvement of Conceptual Site Model and Groundwater Flow Model of Nuclear Power Plants in Korea

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In order to monitor the contamination of groundwater due to unplanned release of radioactive materials and the spread to off-site environments, the nuclear power plants (NPPs) conduct groundwater monitoring program (GWMP) in Korea. The GWMP should be established based on the groundwater flow model reflecting the conceptual site model (CSM) of the NPP's site. In this study, in order to optimize the GWMP, the existing CSM and the groundwater flow model of the domestic NPPs site was updated by reflecting the latest groundwater level.

As part of the CSM improvement, the hydrogeological units were subdivided more detailed from three to six through the review of hydrogeological characteristics of the NPPs site. In addition, major variables that affect groundwater flow, such as water conductivity, have been updated.

The groundwater flow model was revised overall as the CSM was improved. In particular, the excavation depth of the structure and backfill area generated during the construction stage of the NPP structures was accurately reflected, and the drainage boundary conditions were realistically reflected. To verify the revised groundwater flow model, steady-state correction was performed using the groundwater level measured in April, 2021. As a results of the steady-state correction, the standard error of estimate, root mean square (RMS), normalized RMS, and the correlation coefficient were 0.32 m, 1.692 m, 5.608%, and 0.964, respectively. This means that the groundwater flow model is reasonably constructed.

The CSM and groundwater flow model improved in this study will be used to optimize the monitoring location of groundwater in NPPs.

Keywords: Conceptual site model, Groundwater flow model, Groundwater monitoring program, Nuclear power plant

The Sources of Unplanned Releases to Groundwater in the US NPPs

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There are many Systems, Structures, and Components (SSCs) in Nuclear Power Plants (NPPs). The systems include radiological waste treatment system, spent fuel pool cooling, emergency core cooling systems, etc. The structures include reactor building, piping vaults, radioactive waste storage facilities, etc. The components include valves, pumps, piping segments, etc. Radionuclides exist in some of these SSCs and unplanned release may occur when leaks or spills from them. And also Work Practice (WP) is another reason of unplanned release in NPPs. The WP is defined as an action taken by individuals during maintenance, operational or support activities, which could result in or prevent a spill or leak of a radioactive solid, liquid or gas that has a credible mechanism for contamination of groundwater.

According to the results of the Electric Power Research Institute (EPRI) survey, a total of 323 unplanned release event occurred at US NPPs from 1970 to 2014. Among them, 219 events were counted to have occurred at pressurized water reactors (PWRs). In addition, it was confirmed that 41 of the 44 PWR sites (about 93%) in the US, operated at the time of the survey period, had experienced at least one unplanned release events of licensed material which impacted groundwater. This means that the US PWR sites have experienced an average of approximately 5 unplanned release event per site.

The source with the most unplanned releases, including SSCs and WP, was miscellaneous systems with a percentage of about 33% (72 events). Miscellaneous systems include pipes, and it was confirmed that unplanned releases mainly occurred in pipes such as the main steam system, condensate and feed-water system, and emergency core cooling system. And the percentage was high in the order of WPs (21%, 45 events), radioactive effluents (20%, 43 events), refueling water storage (8%, 17 events), radioactive waste/material operations (7%, 16 events), spent fuel storage (5%, 12 events), unknown (4%, 9 events), and structures (2%, 5 events).

The history of the unplanned release of the US NPPs will be considered when revising major SSCs in the domestic NPP groundwater monitoring program.

Keywords: SSC, Work practice, Unplanned release, Groundwater contamination

Analysis of Selection Schemes for Prototyping the Protection and Evacuation Items in Complex Radioactive Disaster

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A radioactive complex disaster refers to a situation in which large-scale natural and social disasters such as earthquakes, fires, and chemical incidents occur in addition to nuclear power plants accident simultaneously. For the safe evacuation of residents, protective equipment must be prepared appropriately and sufficiently. We presented effectiveness, sufficiency, and versatility as selection criteria for choosing the essential protective equipment. The results of the survey of residents and local public officials suggested that masks, radioactivity meters, protective buses, and air purifiers have high priorities. Finally, through consultation with central government officials, masks, measuring instruments, and adsorption filters in protective buses were presented as prototypes. These items can effectively protect the people even in the event of a radioactive complex disaster.

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Keywords: Complex radioactive incidents, Protection items, Decision criteria

Sabotage Scenario Radiation Sensitivity Evaluation for Domestic Spent Nuclear Fuel Transport Storage Cask

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To rationalize the protection of spent nuclear fuel transport storage cask, we intend to investigate the status of domestic and foreign safety regulations and related technologies to develop sabotage scenarios and analyze the protection performance and radiation impact of transport storage cask. It is essential to conduct an aircraft collision safety evaluation on spent nuclear fuel transportation and storage casks in Korea due to changes in laws and regulations related to nuclear power plant design and demand for enhanced safety. Domestic and foreign research on the protection performance of spent nuclear fuel transport storage cask was based on 9.11 events, and the results of all studies show that the speed of the aircraft and leakage of nuclear materials are insignificant. The Sandia National Laboratory (SNL) calculates Aerosol emissions from spent fuel damage in the event of sabotage and calculates Source Term based on the Durbin-Luna model. In this paper, radiation sensitivity analysis was performed due to damage to the carrier according to the size of the accident, assuming that there was a hole enough to basket from the external shell among the collision scenarios identified for domestic cask models.

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Keywords: Spent fuel transportation cask, Sabotage, Physical protection, Radiation effect

Comparison of Source Term Assessment of the Release of Severe Accident Before and After the Fukushima Accident

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Attention has been paid to the source term released after Chernobyl and Three Mile Island (TMI), which were the representative accidents of nuclear power plants, and has been studied several times in order to predict and evaluate radiation source term, which can be released in the event of a virtual accident.

In particular, the impact of the accident was assessed on the basis of Deterministic Safety Analysis (DSA) and after the WASH-1400, the technology of the Probabilistic Safety Assessment (PSA) was introduced, supplementing safety by taking into account the existence of uncertainty. After the Fukushima accident, a SOARCA report was published to evaluate the specific classification of each type of accident, the realistic progress of the accident, and the leakage of radioactive materials.

In this paper, the evaluation methodology and results of the source term of severe accident before and after the Fukushima accident were compared, and the evaluation methods applied to domestic nuclear power plants were compared. Prior to the Fukushima accident, the behavior of the accident and source term were evaluated for Loss of Coolant Accident (LOCA), which led to design based accidents, Total Loss of Feed Water (TLOFW) followed by Station Blackout (SBO) the results were compared to Chernobyl and TMI based on the resulting data to evaluate safety and reliability. After the Fukushima accident, the Interfacing System Loss of Coolant Accident (ISLOCA) and the Steam Generator Tube Rupture (SGTR), which is classified as containment's bypass accident, were included for predictive assessment. This is due to the analysis that the risk of cancer and early mortality are affected. MACST facilities and strategies were added to domestic nuclear power plants, and accidents with a high core damage frequency were mainly interpreted. In addition, source term was evaluated with the addition of a Basement Melt-Through (BMT) accident that had not previously been considered as a focus. As a result of the comparison of source term evaluation, accidents can be caused by a number of unidentified problems, and Korea's experience on Level 2 and 3 has not been accumulated, making it difficult to predict the results of source term evaluation or lack of reliability.

Keywords: Source term, Severe accident, Fukushima, MACST, SOARCA

Performance Evaluation by Measurement Conditions for Wide Area Site Exploration Design

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Due to the Fukushima nuclear accident, a large amount of radioactive material was released into the atmosphere, and consequently, it spread over a wide area and was deposited into the soil. As a result of this, a wide area of radioactive contamination site was created. Due to the contaminated site, the need for research on various exploration platforms for efficient situation management and field response is being emphasized.

Backpack-type radiation survey & monitoring equipment is useful for creating a contamination maps containing information such as Dose Rate, Radionuclide, Activity Concentration accompanied by spatial analysis when performing a Scan Survey that moves with a backpack on a wide area site. Contamination maps are based on accurate radiological characteristic information. However, there is a problem in that the accuracy of the evaluation results is lowered due to changes in environment conditions or the variability of the dose rate and counting rate during scan survey. This problem should be solved by applying the influence of each variable to the underlying data. However, prior to this, it is most important to prepare the base underlying first. And this can be obtained through evaluation of detection performance through static survey. Therefore, in this study, the change in detection efficiency for the measurement height and radius of the backpack-type radiation survey & monitoring equipment based on the 3"×3" NaI(Ti) detector was evaluated.

First, the height of the source and Backpack-type radiation survey & monitoring equipment was evaluated from 0 cm to 1 m, which is the height of the soil and detector when an adult male wears a backpack. The experiment was conducted using the ^{137}Cs (383 kBq) point source, which is a nuclide mainly detected at the contaminated site. The measurement time was measured five times per one minute, considering that it was backpack-type equipment and a future scan survey. In addition, in order to evaluate the detection radius, the measurement was performed by changing the measurement distance up to 5 m at intervals of 50 cm. As a result of evaluating the detection performance of the backpack-type radiation surveys & monitoring equipment, it was confirmed that increasing the detection height and radius reduces the count rate in the form of an exponential function. In addition, it was confirmed that the detected radius varies depending on the height. Based on these results, we plan to conduct additional research to understand the scan survey and its sensitivity to various factors. Through this, the company plans to develop various models for exploring the site by improving the accuracy of backpack-type radiation surveys & monitoring equipment.

Keywords: Backpack, Contamination, Site, NaI(Ti), Counting efficiency

Introduction of Radiation Measurement Situation Virtual Reality Program

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“Radiation measurement situation virtual reality program” is a three-dimensional modeling program. It is a program that virtually implements the measurement situation by applying visualization techniques such as animation effect charts, and effectively delivers gamma radiation energy spectrum data. This program was developed to respond to various measurement situations by visualizing major analysis objects such as radiation detectors, radioactive waste drums, radioactive building structures, and the ground surface as a 3D model. User-friendliness was secured by supporting various control functions such as distance, size, and angle while checking the three-dimensionally produced detector and analysis target in a virtual space. By using high-resolution photos obtained through 360-degree shooting, a virtual space was implemented to approximate the actual situation, such as the ground surface measurement. In addition, data communication safety was secured so that a large number of users could use it through a local area network in consideration of the actual operating environment.

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Keywords: Virtual reality, Radiation measurement, 3D modeling, Gamma radiation energy spectrum

Analysis of Effect of the Shape Factor on External Dose Rate in RADTRAN by Comparison With MCNP

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RADTRAN is a code that assesses the radiation risk of radioactive material transportation. RADTRAN assumes that the package is a point source or a line source regardless of package type and corrects the external dose rate using a shape factor which depends on the critical dimension of the package. However, the external dose rate calculated using a shape factor may be different from the actual external dose rate. Therefore, it is necessary to analyze the effect of the shape factor on the external dose rate. In this study, the effect of the shape factor on the external dose rate in RADTRAN was analyzed by comparison with MCNP.

This study analyzed change in external dose rate depending on the distance from the package and the critical dimension. The distance from the package was in the range of 1–800 m. The shape of the package was assumed to be cylindrical with a radius of 1 m, and the critical dimensions of the package were assumed to be 2, 4, and 8 m. Attenuation and build-up in the air were not considered to consider only the effect on the shape factor. When simulating the exposure situation using MCNP, the package was assumed to be a volume source, and flux by distance from the package was calculated using F5 tally. The dose rate at 1 m from the package was normalized to $2 \text{ mSv} \cdot \text{hr}^{-1}$.

As a result of the analysis, the external dose rates of the package were higher in RADTRAN than in MCNP. For the critical dimension of 2, 4, and 8 m, when the distance from package is 1–10 m, the RADTRAN was 1.83, 4.08, and 5.27 times higher on average than MCNP, respectively. And when the distance from the package was 10–100 m and 100–800 m, RADTRAN was 1.10, 2.02, 3.01 times and 1.04, 1.92, 2.43 times higher than MCNP, respectively. It was found that the larger the distance from the package is and the smaller the critical dimension of the package is, the less conservatively RADTRAN assessed. It is because the shape of the package gets closer to the point source as the distance from the package increases, and the shape factor decreases as the critical dimension of the package decreases. The result of this study can be used as the basis for radiation risk assessment when transporting radioactive materials.

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Keywords: Radiation risk assessment, External dose rate, Shape factor, RADTRAN, MCNP

Revisit of LLD Requirements for Airborne Discharges From PWRs in Line With Effluent Concentration Limit

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Radioactive effluent discharged from the nuclear power plant (NPP) during normal operation is controlled by the discharge limit in terms of radioactivity concentration ($\text{Bq}\cdot\text{m}^{-3}$) and dose constraints in Korea. To ensure compliance with discharge limits of effluents, the licensee operates radioactive effluent monitoring systems in each discharge point to detect radioactivity and control discharge. The predetermined regulatory requirements of analytical sensitivities for sampling devices in the monitoring system are established in various countries to guarantee the performance of the monitoring systems. In Korea, Lower Limits of Detection (LLD) are selected as the regulatory requirements and adopted from the United States Nuclear Regulatory Commission (USNRC) NUREG-1301. The International Atomic Energy Agency stated that the detection limits have to be low enough (e.g., less than 1% of discharge limits) to safely demonstrate compliance with the discharge limits. However, no technical background of LLD has been explained regarding the compliance with discharge limits in Korea. Thus, it is necessary to analyze the compatibility of discharge limits and detection limits.

The USNRC Regulatory Guide 1.21 has stated the risk-informed approach for effluent control by identifying the principal radionuclides whose radiological impact is more than 1% of discharge limits. In 2017, Cheong proposed the methodology and derived risk-based detection limits for liquid effluents from Korean NPPs. In 2019, Choi derived risk-based detection limits for liquid and gaseous effluents based on APR 1400 Design Control Document (DCD). The methodology of those studies can derive the detection limit for each principal radionuclide that is comparable to 1% to 10% of discharge limits. However, the previous study based on APR 1400 DCD was for the discharge limits of the US and didn't consider the multiple discharge points in the reactor.

Therefore, this study preliminarily derived the risk-based detection limits consistent with Korean Effluent Concentration Limits for gaseous effluents reflecting the characteristics of each discharge point. Also, this study confirmed the validity of risk-based detection limits and current LLD. This study is expected to be basic research for detection limits of Korean NPPs in line with international safety standards.

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Keywords: Lower limits of detection, Radioactive effluents, Discharge limits, Radiation monitoring system, Risk-informed approach

A Study on Determining the Detection Efficiency of Waste Drums for Safety Management of Radiation Worksites

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It is essential to provide a safe working environment for radiation workers. At a research reactor decommissioning site in Seoul (KRR1 & KRR2), radioactive waste drum disposal work is in progress. Before performing radiation work, it is necessary to determine the radioactivity of the waste drum to ensure safety. In this reason, we conducted a study to determine the detection efficiency of waste drums using the EXVol code.

Determination of the full energy absorption peak efficiency (detection efficiency) is one of the important processes of the gamma-ray activation analysis. For the large voluminous gamma-ray sources like waste drum, the geometrical and attenuation effect should be considered. EXVol (Efficiency calculator for eXtended Voluminous source) code is a detection efficiency calculation code using the effective solid angle method. EXVol can calculate both coaxial and asymmetric structure. In addition, the introduction of a collimator made it possible to reduce the radiation intensity of a high radiation source. And it is possible to determine the precise detection efficiency according to the energy of a gamma ray at a specific position of the volume source. To verify the performance of the EXVol, a high resolution gamma spectroscopy system was constructed and measurement and analysis were performed. Measurements were performed on coaxial, asymmetric and collimated structures with standard point source, standard 1 L liquid volume source and HPGe detector. The measured results were compared with the calculation results of EXVol. The relative deviation of the measurement and calculation in the coaxial and asymmetric structures was 10%, and that of the collimation structure was 20%. Results can be available in analysis of waste drums' radioactivity determination at a specific position.

Keywords: Radiation safety management, Radioactive waste, Detection efficiency, Collimation geometry

~~PWR Fresh Fuel Shipping Package Development~~

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Canceled Poster

Development of Standard Radiation Sources for Verification of Classification System of Nuclear Decommissioning Waste

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This study is about the production of radiation sources of simulated concrete and soil reference materials to verify the validity of the quality establishment and measurement of the detector (HPGe) of the radioactive soil and concrete waste classification system, which is being developed to quickly and accurately classify nuclear decommissioning waste. Specific activity of gamma nucleus among radioactive wastes is evaluated using gamma spectroscopy. At this time, in order to verify the validity and reliability of measuring equipment, it shall be a standardized substance of the same medium as nuclear decommissioning waste (chemical ingredients, particles, density, etc.) in order to correct the energy and efficiency of gamma nuclide analysis equipment.

The CRM used for the detector's energy correction used a 1 L Marinelli beaker standard correctional radiation source consisting of 10 radioactive isotopes. In order to correct efficiency, in accordance with the production and certification process of the Korea Standards and Research Institute, it has produced artificial simulated radioactive concrete similar to nuclear decommissioning waste (30% for cement, 60% for regulation and 10% for bentonite). The radioactive homogeneity of the simulated concrete reference materials was evaluated using dispersion analysis (ANOVA) in accordance with ISO Guide 35, while ¹³⁷Cs and ⁶⁰Co of concrete reference materials were able to obtain homogeneous measurements both in and between bottles.

The self-absorption rate of the simulated concrete reference material was determined by the MCNP computer simulation measurement method, and the self-absorption correction coefficients of ¹³⁷Cs and ⁶⁰Co were assessed at 0.995 and 0.996, respectively, and the standard value for the radiation of the simulated concrete reference material was calculated on the weighted average of the measurements of 20 samples. The uncertainty about the reference value was calculated by combining measurement uncertainty (Type B evaluation), bottle to bottle standard deviation, and uncertainty within bottle by modifying the formula suggested in ISO Guide 35.

The concentration of ¹³⁷Cs and ⁶⁰Co of reference materials was divided into high-speed measurement mode and precision measurement mode in consideration of the self-disposal standard. The reference value and uncertainty of expansion among reference materials for high-speed measurement mode were rated at $1,032.7 \pm 64.0 \text{ Bq}\cdot\text{kg}^{-1}$ and $1,083.7 \text{ Bq}\cdot\text{kg}^{-1}$, respectively. The standard value and expansion uncertainty for ¹³⁷Cs and ⁶⁰Co among reference materials for precision measurement mode were rated at $113.7 \pm 10.0 \text{ Bq}\cdot\text{kg}^{-1}$ and $122.3 \pm 10.3 \text{ Bq}\cdot\text{kg}^{-1}$, respectively.

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Keywords: Decommissioning, Radioactive source, Cs-137 and Co-60, Self-absorption factor

Evaluation of Shielding Thickness of Transport Cask for Transporting Co-60 Source Rods Produced Using PWRs

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The Co-60 is a radioactive material widely used in domestic and foreign medical, industrial, health and research fields. Currently, world market for the Co-60 is about 80 MCi/yr and is expected to grow to 150 MCi/yr by 2025. For the Co-60, Nordion of Canada occupies about 80% of the world market.

In the case of Korea, a small amount of sources with relatively low radioactivity intensity are produced

using research reactors, but most of the Co-60 is entirely dependent on imports. Accordingly, although the technical feasibility of the Co-60 production technology using the PHWR was evaluated, it was evaluated as a negative result on the additional construction of a hot cell, core management, safety analysis and economic feasibility. Canada, the main producer of the Co-60, is also conducting research on the Co-60 production technology using PWR with GE-Hitachi and Westinghouse as the number of PHWR is expected to decrease.

In Korea, it is necessary to preoccupy the Co-60 production technology and auxiliary technology using the PWR by utilizing excellent technology, and active research is being conducted to secure unique nuclear power technology that does not depend on foreign countries. Therefore, in this study, the thickness and weight of the radioactive shielding required for handling (transport) of Co-60 produced using the PWR were calculated.

Keywords: Co-60, Transportation, Shielding

Analysis of Storage Procedure by Method of Spent Nuclear Fuel Dry Storage

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In Korea, research on the introduction of dry storage facility is being conducted as an alternative to saturation of temporary storage facilities for spent nuclear fuel. The introduction of dry storage facilities requires a radiological impact assessment on the workers of the facility, and for this, an appropriate exposure scenario must be derived through work procedure analysis. In this study, the procedure for storing spent nuclear fuel in dry storage facilities was analyzed based on the case of evaluating the radiological impact of workers in dry storage facilities abroad.

We investigated cases of radiological impact assessment on workers at on-site dry storage facilities by PNNL, Dominion, and P. F. Weck. PNNL and Dominion analyzed the storage work procedure of the VSC (Vertical Storage Cask) method using CASTOR V/21, TN-32, respectively, and conducted a radiological impact assessment. P. F. Weck analyzed the storage work procedure of various spent nuclear fuel casks for VSC and HSM (Horizontal Storage Module), conducted a radiological impact assessment. As a result of comparing the procedure for storing spent nuclear fuel by case, it was found that the storage procedure was determined by the storage method and the cask type. In the case of VSC method, canister-type casks and basket-type casks are used, and the storage procedure are partially different according to each. Canister-type cask requires repackaging from transfer overpack to storage overpack, but basket-type cask doesn't require that procedure. In the case of the HSM method, only the canister type cask was found to be used. However, the storage procedure was different depending on the type of HSM system. Depending on the type of HSM system, the necessity of cask for on-site transport was different.

In this study, we investigated and analyzed the work procedure according to the storage method of dry storage facilities abroad. It was found that the dry storage procedure of spent nuclear fuel different according to the storage method and type of cask. The results of this study can be used as basic when deriving the exposure scenario for spent nuclear fuel dry storage workers suitable for the domestic situation.

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Keywords: Spent nuclear fuel, Dry storage, Exposure scenario, Work procedure, Radiological impact

Sensitivity Analysis of Contaminated Zone for Clearance of Decommissioning Concrete Waste Subject to Landfill

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The dose calculation for workers and residents after landfill closure and sensitivity analysis of contaminated zone was performed. It is expected that 80,000 drums of radioactive waste will be generated during the decommissioning of a nuclear power plant. About 20% of the radioactive waste is concrete waste. Among them, bioshield concrete which has clearance level can be considered for a subject of self-disposal. It is essential for radiation safety to confirm whether the self-disposal criterion limit is satisfied. The code used for the self-disposal evaluation is the RESRAD code, which enables the calculation of guidelines for radioactive materials in the soil that meet the dose criteria and the exposure dose to workers and residents.

Through the RESRAD code, the expected exposure dose of landfill workers and residents after landfill closure was calculated to determine whether the self-disposal criterion limit was satisfied. The assumptions for the evaluation are follows: The specific radioactivity of each radionuclide was applied based on the time of cooling for 8.5 years, and the thickness of contaminated zone is 2 m which is the default value of RESRAD. As a result of the evaluation, the maximum radiation dose for workers was $2.19 \times 10^{-1} \mu\text{Sv} \cdot \text{yr}^{-1}$, and the maximum radiation dose for residents was $1.18 \times 10^{-2} \mu\text{Sv} \cdot \text{yr}^{-1}$, which satisfied the self-disposal criterion limit ($10 \mu\text{Sv} \cdot \text{yr}^{-1}$).

In addition, although it is more efficient to thicken the contaminated zone in a narrow area, a sensitivity analysis was performed to determine whether there is a significant difference depending on the thickness of contaminated zone. The sensitivity analysis for the thickness of contaminated zone of 1, 1.5, 2, and 2.5 m was performed. As a result, the maximum radiation dose for the thickness of contaminated zone of 1 m was 1.1 times larger for workers and 1.75 times larger for residents than that of 2 m. When the thickness of contaminated zone was 1.5 m, it was 1.03 times larger for workers and 1.25 times larger for residents than when it was 2 m. The dose had a value of 0.98 times for workers and 0.85 times for residents of that of 2.5 m, respectively. The difference seems to be insignificant. Thus, it would be good to set it up according to the actual conditions of the landfill site. Also, it is necessary to study the sensitivity analysis for other variables such as cover depth, erosion rate, etc.

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Keywords: Dose evaluation, Clearance, Concrete waste, Landfill

Comparison of Effective Dose Calculation Results of PHITS and MCNP in 40 MeV Cyclotron

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Korea Institute of Radiological and Medical Sciences provides proton irradiation service of up to 40 MeV using cyclotron. The use of such a cyclotron was approved in advance to satisfy the Nuclear Safety Act, and radiation safety was evaluated in this process. The Monte Carlo method is generally used to evaluate the shielding safety of high-energy accelerators, and MCNP 6.2 was used in the previous evaluation. In this study, in order to verify the results of previous evaluation, the calculation results of MCNP 6.2 and Particle and Heavy Ion Transport code System (PHITS) 3.24 are compared. PHITS is a general-purpose Monte Carlo particle transport simulation code that is used in many studies in the fields of accelerator technology, radiotherapy, space radiation, etc.

In the previous evaluation, the effective dose by neutrons and photons generated by the collision of 40 MeV 20 μ A of protons with a 10.5 mm thick beryllium target was evaluated, and in this study, this was reproduced with PHITS. As the radiation exposure evaluation for the user or public is evaluated based on the radiation dose and energy distribution generated around the target, the effective dose and energy distribution received by the water phantom with a radius of 1 cm on the front, side, and back of the target were calculated. T-Track, a tally of PHITS, was used to calculate effective dose, which is similar to F4 tally of MCNP 6.2 using a dose conversion factor. For the dose conversion factor, the value suggested as AP irradiation in Publication 103 was used.

As a result of the calculation, the effective dose by neutrons at the front, side and back of the target was 1.42×10^5 , 2.09×10^4 , and 1.39×10^4 mSv $\cdot$ h⁻¹, respectively, which was similar to 2.00×10^5 , 1.84×10^4 , and 2.59×10^4 calculated using F4 tally in MCNP. Moreover, the results of calculating the effective dose by photons using PHITS were 4.81×10 , 3.10×10 , and 2.66×10 , respectively, and the results of calculating MCNP were 4.49×10^2 , 6.45×10 , and 9.64×10 . The average energies of neutrons were 11.2, 0.69, and 0.31 MeV when calculated by PHITS, respectively, and 13.8, 7.8, and 4.6 when calculated by MCNP. Moreover, the average energies of photons were 1.98, 0.98, and 0.86 when calculated by PHITS, respectively, and 3.9, 3.2, and 2.6 when calculated by MCNP.

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Keywords: PHITS, Monte carlo, Radiation safety, Dose calculation, MCNP

Dose Evaluation of Workers Transporting Spent Resin Drums Treated From the Spent Resin Mixture Treatment Facility

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The dose was evaluated for the workers transporting the spent resin drums from a spent resin mixture treatment facility. The treatment technology of spent resin mixture waste based on microwave was developed to compensate for the shortcoming of the existing one. The mechanism of the facility for the treatment is divided into separation, desorption, condensation and adsorption process. The treated spent resin that has passed through the microwave reactor flows into the spent resin storage tank. As the treatment time elapses, if spent resin accumulates in the spent resin storage tank, it is moved to the drum of the volume of 200 L. The drum must be moved by the worker, in which case radiation exposure to the drum transport worker occurs. It requires the dose evaluation for drum transport workers in terms of radiation safety. Dose evaluation was performed in consideration of the change in the composition ratio and weight of the spent resin mixture, where the working time for transportation was considered from 10 to 120 minutes in 10-minute increment. In the case of 100 kg of the spent resin mixture, the dose range was derived as $4.62 \times 10^{-3} - 5.90 \times 10^{-2}$ mSv for the 100 kg of spent resin, $4.72 \times 10^{-3} - 5.58 \times 10^{-2}$ mSv for the 80 kg of spent resin and 20 kg of zeolite and activated carbon, and $5.38 \times 10^{-3} - 6.32 \times 10^{-2}$ mSv for the 60 kg of spent resin and 40 kg of zeolite and activated carbon. In the case of 150 kg of the spent resin mixture, the dose range was derived as $6.83 \times 10^{-3} - 8.20 \times 10^{-2}$ mSv for the 150 kg of spent resin, $7.13 \times 10^{-3} - 8.22 \times 10^{-2}$ mSv for the 120 kg of spent resin and 30 kg of zeolite and activated carbon, and $8.28 \times 10^{-3} - 8.86 \times 10^{-2}$ mSv for the 90 kg of spent resin and 60 kg of zeolite and activated carbon. The estimated maximum doses for each weight (100 kg and 150 kg of mixture) were confirmed to be $3.16 \times 10^{-1}\%$ and $4.43 \times 10^{-1}\%$ of the annual average dose limit of 20 mSv for radiation workers.

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Keywords: Dose evaluation, Spent resin mixture, Treatment facility, Drum transportation

Review on Off-site Dose Calculation Trend

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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 evaluates the dose of residents around nuclear power plants using the K-DOSE60 program that reflects ICRP-60. K-DOSE60 calculates the expected exposure dose for residents by modifying the atmospheric dispersion and deposition factors evaluation module (XOQDOQ), gaseous effluent evaluation module (GASDOS) and liquid effluent evaluation module (LIQDOS) developed by the US NRC.

The current evaluation program is the Bounding Assessments method, which evaluates under the assumption that residents reside at the exclusion area boundary (EAB), and has a disadvantage in that the estimated exposure dose is evaluated too conservatively. In the EPRI, instead of the conservative method that is conventionally performed for the residents' dose evaluation method, a plan to improve the accuracy of the dose evaluation reflecting the site characteristics was reviewed. In addition, improvements were derived through the review of NPPs operation status, experience cases and the latest technology.

Keywords: ODCM, Liquid effluent, Gaseous effluent, K-DOSE60

Analysis of the Elevation Database Around the Nuclear Power Plant for Radiological Emergency Preparedness

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Once a radioactive material is released from the nuclear power plant (NPP) by accident, it is necessary to understand the behavior of radioactive plume to protect residents adequately. For this, it is essential to measure the radiation dose rate around NPPs at important locations. Our previous study developed a movable radiation detector that can be installed quickly in an accident to measure gamma dose rate in areas where environmental radiation monitoring system is not installed. The data measured by the detector are transmitted to the server in real-time through LoRA wireless communications. There are two methods to use LoRA communications; one is self-network, and the other is the network provided by the mobile carrier. A signal receiver, called a gateway, should be equipped near the installation location of radiation detectors to use a self-network without using the mobile carrier's system. In other words, the movable radiation detectors we made can function if there should be any gateway near them. The distance capable of communication between gateway and detector is about 8 km in an open area without significant obstacles. Korea has many significant obstacles, such as mountains around most NPPs. Thus, the gateways could be installed in the proper position before the accident to operate the movable radiation detectors without problems. If the gateway is located at a high position like a mountain top, it could cover a wide area. In this study, the elevation database in the area around the NPPs was collected and analyzed to determine where gateways should be installed. The analysis range is limited in the urgent protective action planning zone. The optimization was also performed to minimize the number of gateways.

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Keywords: LoRA wireless communications, Movable radiation detector, Radioactive plume, Radiological emergency preparedness, Optimization

Calculation of the Characteristic Limits for a Noble Gas Monitor Based on the ISO 11929 Publication

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There are three questions arise in radioactivity measurements: (1) Dose the measured value originate from the radioactivity being present in the sample? (2) Is the measurement procedure suitable for the intended measurement purpose with respect to the requirements? (3) What is a range of values fairly sure our true value lies in with a specified probability? These three questions are answered by determining characteristic limits (decision threshold, detection limit and limits of the coverage interval), which are widely used as part of quality assurance in radioactivity measurements.

In the past, numerous papers have focused on the questions in different ways, and have drawn a variety of conclusions about the meaning of the different characteristic limits using various terms and symbols. In recent years, substantial efforts were made in order to obtain a systematic and unified way to calculate and express these limits. As a result, the ISO 11929 Series which specify a procedure for calculation of the characteristic limits have been developed.

This paper is focused on the calculations of characteristic limits for noble gas monitor (NGM204 monitor) that offers the continuous measurements of radioactive noble gases discharged from the stacks of the HANARO reactor facility. The calculations are based on the standard ISO 11929 as well as the traditional formulas provided by NUREG 1576, ANSI N42-18, ANSI N42-17 and DIN 25482. A comparison is made among the results obtained from the formulas given in each literature.

Keywords: Decision threshold, Detection limit, Limits of the coverage interval, ISO 11929, Noble gas

The Status of Groundwater Monitoring of Nuclear Power Plants in the US

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In 2005, groundwater contamination due to unplanned releases of radioactive materials from the US. Nuclear Power Plants (NPPs) such as Braidwood and Indian Point was confirmed. The following year, in 2006, The Nuclear Regulatory Commission (NRC) established a task force team to investigate the history of unplanned release of all NPP in the US. As a results 217 events of unplanned release including leaks and spills were identified in the US NPPs.

The NRC regulates the radioactivity concentration of off-site groundwater by setting a reporting levels (RLs), and if exceeds the RLs, the licensee must report within 30 days. When the off-site groundwater is used as drinking water or non-drinking water, the RLs for tritium in groundwater are $740 \text{ Bq}\cdot\text{L}^{-1}$ or $1,110 \text{ Bq}\cdot\text{L}^{-1}$, respectively. Whereas the NRC does not set the RLs for on-site groundwater.

The Nuclear Energy Institute (NEI) issued the guidance document “Industry groundwater protection initiative” NEI 07-07 in 2007. And the members of the NEI promised with regulatory body and local governments to implement groundwater monitoring/protection program according to the NEI 07-07. The document states that when the on-site groundwater is used as drinking water, the RL ($740 \text{ Bq}\cdot\text{L}^{-1}$) for off-site groundwater will be applied and the licensee voluntarily reported to the NRC. And also, NPPs are setting the Investigation Level (IL) below the RP and the IL is various among the NPPs. The IL is the standard by which detailed investigations are implemented when the level (radioactivity concentration) is exceeded.

Keywords: Reporting level, Investigation level, On-site groundwater, Groundwater monitoring

Effect of Concrete on Thermal Neutron in the Application of Neutron Tomographic Technique Using He-4 Gas Scintillation Detector

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For spent nuclear fuel transferred to dry storage facilities, it is difficult to apply safeguards approaches and long-term integrity verification due to the structural characteristics of the facility. There is a need to check the integrity of the nuclear fuel assembly before transferring it to a dry storage facility and are need to provide information on whether there are any defects. At the Korea Institute of Nuclear Nonproliferation and Control, as a non-destructive testing technology for ensuring Continuity of Knowledge (CoK) of the dry storage facilities, a methodology for reconstructing images by neutron tomographic technique from spent nuclear fuel using a He-4 gas scintillation detector was presented. It is thought that the He-4 gas scintillation detector-based technology can be used to verify the defect of the nuclear fuel assembly. This methodology must be accompanied by accurate neutron measurements. The place where the technique was conducted is surrounded by a concrete wall. Concrete contains water molecules, which can affect neutron measurements. In this study, reconstruction images based on neutron measurements and MCNP simulations are compared to verify the effects of the concrete. Neutron measurements were performed by measuring Cf-252 neutron sources in a 1/10 lab-scale TN-32 cask with six He-4 gas scintillation detectors as an array. Neutron sources are fixed at each point in the cask, and the He-4 detector array is rotated from 0° to 360° at 10° intervals to reconstruct the image using the filtered back-projection (FBP) method. Also, in MCNP reconstructed images, there are two versions depending on whether concrete wall. The source image and ring shape were found in the measurement-based thermal neutron reconstruction image, which was similar to the simulation image that considering the concrete effects. On the other hand, in the simulation reconstruction image without the concrete, only the shape of the source was found. Thus, the effect of concrete should be considered when performing the neutron tomographic techniques using He-4 gas scintillation detectors.

Acknowledgements

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Keywords: Spent nuclear fuel, Safeguards, He-4 gas scintillation detector, Thermal neutron

7분과

방사화학(Oral)

Computational Radiochemistry Framework for Understanding of Nuclear Fuel Properties Using First-principles Integrated With Machine Learning Approach

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The success of machine learning approach to identify key correlation in large database is critically controlled by the reliability and accuracy of the data. Here, we demonstrate that rigorous material properties of radioactive nuclear fuels can be obtained by integrated approach of first principles calculations and the machine learning approach. The reliable database is established by density functional theory and molecular dynamics simulations, which is the input of the machine learning to analyze any correlation among the database. The outcomes are applied to evaluate thermodynamic, kinetic and electrochemical properties, which plays a key role for safe management of spent nuclear fuels.

Acknowledgements

This work was supported by the Nuclear R&D Program of the National Research Foundation of Korea (NRF) grant funded by the Ministry of Science and ICT (NRF-2016M2B2B1945254).

Keywords: First principles, Machine learning, Spent nuclear fuel, Radiochemistry, Database

Impact of dopants on $^{99}\text{Tc(IV)}$ Mobility in Novel Waste Form *ab initio* Molecular Dynamics Simulations

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^{99}Tc is a long-lived radioactive fission product whose mobility in the subsurface is largely governed by its oxidation state. Migration of Tc from a waste repository may be prevented by immobilizing Tc(IV) in durable glass forms. Thus, efficient incorporation and high retention of Tc by glasses is very important for radioactive waste management and environmental remediation. Tc(IV) oxidation to highly volatile Tc(VII) (TcO_4^-) at glass vitrification temperatures results in poor Tc retention in the final waste glass. Retention of Tc in the glass is generally improved by reducing conditions since Tc(IV) is not volatile. However, experiments with Tc-magnetite under high temperature and oxic conditions showed re-oxidation of Tc(IV) to volatile pertechnetate. To understand this phenomenon, we employed *ab initio* molecular dynamics simulations to study structural and electronic properties of Tc-incorporation in magnetite. The study provides the ramification of high temperature on Tc mobilization and changes of Tc retention in magnetite. Theory predictions also indicated enhanced Tc retention in the presence of 1st row transition metal dopants in Tc-incorporated magnetite that was confirmed by experiments. Furthermore, the overwhelming concentration of Cr(VI) co-mingled with Tc(VII) within the secondary waste form where Cr(VI) is more readily reduced than Tc(VII) by reduction potential metrics. Experiments with Tc-magnetite under high temperature and oxic conditions showed re-oxidation of Tc(IV) to volatile pertechnetate. Experiments also showed that magnetite transforms to maghematite resulting in disproportionation and re-oxidation of Tc. This transformation can be suppressed through incorporation of trace elements such as Co, Ni, Zn into magnetite forming spinel.

In this talk, I will present results from *ab initio* molecular dynamics simulations and experiments on the structural and electronic properties of Tc and/or Cr incorporation in magnetite. I will also discuss the temperature effect on Tc/Cr mobilization and changes of Tc/Cr retention in magnetite. In addition, I will discuss the effect of Tc/Cr concentration and 1st row transition metal dopants on their incorporation in magnetite.

Keywords: Tc, Glass waste form, Radioactive waste management, Immobility, MD simulation, Radionuclide retention

Machine Learning Approach for Simultaneous Determination of Counting Efficiencies of Strontium-90 and Strontium-89

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Radioactivity of radiostrontiums, Sr-89 and Sr-90, which are both pure beta-emitters, are generally measured via Cherenkov counting. However, the determination of Cherenkov counting efficiencies of radiostrontiums requires a complicated procedure due to the presence of Y-90 (also a pure beta-emitter) which is the daughter nuclide of Sr-90. In this study, we have developed a machine learning approach using a linear regression model which allows an easier and simultaneous determination of the Cherenkov counting efficiencies of the radiostrontiums. The linear regression model was employed because total net Cherenkov count (C_t) from the three beta-emitters at time t after the separation of Y-90, can be expressed as a linear combination of their respective time-varying radioactivities with their respective coefficients (parameters) being their counting efficiencies: $C_t = \varepsilon_{\text{Sr-90}}[A_{\text{Sr-90}} \cdot \exp(-\lambda_{\text{Sr-90}} \cdot t)] + \varepsilon_{\text{Sr-89}}[A_{\text{Sr-89}} \cdot \exp(-\lambda_{\text{Sr-89}} \cdot t)] + \varepsilon_{\text{Y-90}}[A_{\text{Sr-90}} \cdot \exp(1 - \lambda_{\text{Sr-90}} \cdot t)]$, where ε is a counting efficiency, A is an initial activity, λ is a decay constant and t is time after the separation of Y-90. Thus, if we train the model with multiple Cherenkov counts measured from the three beta emitters, then we can obtain their estimates for counting efficiencies (so-called parameters) straightforward. For this, the model has been trained by two methods: Ordinary Least Squares (OLS) and Bayesian linear regression (BLR), for which two software packages, PyMC3 and Stan were employed to compare their performances. The results showed that the accuracy of the OLS was worse than that of the BLR. Particularly, the counting efficiency of Sr-90 was estimated to be smaller than 0, which is an unrealistic value. On the other hand, the estimates of the BLR gave realistic values which are close to the true values. Additionally, the BLR was able to provide a distribution for each counting efficiency (so-called “posterior”) from which various types of inference can be made including median and credible interval in the Bayesian statistics which is analogous to, but different from confidence interval in the Frequentist statistics. In the results of the BLR, the Stan package gave more accurate estimates than the PyMC3 package. Therefore, it is expected that counting efficiencies of the radiostrontiums including radioyttrium can be determined at the same time, more easily and accurately, by using the BLR with the Stan package and that the activities of radiostrontium also can be determined more easily by using the BLR if we know their counting efficiencies in advance. It is worth noting that the usage of the linear regression model in this study was different from the usual one where the trained model is used to predict a response value (count) from a set of unseen regressor values (activities).

Keywords: Radiostrontium, Counting efficiency, Machine learning, Bayesian linear regression, PyMC3, Stan

Ionic Diffusion Oxidation Model of Uranium - A Revisit Study

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After the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) was signed, Korea is undergoing nuclear inspection by the International Atomic Energy Agency (IAEA) as a non-nuclear-armed state. By the inspection, nuclear material measurement and management have been carried out according to safety measures. Uranium dioxide, a major component of nuclear fuel, is a material that naturally oxidizes at room temperature, yielding a volume change. In this case, it will have an impact on the management of nuclear material measurement, and a model for predicting this will be required. At room temperature, an oxide film is grown by oxygen diffusion on the surface of uranium dioxide, and if the thickness of the oxide film is predicted based on this, the volume change of uranium dioxide can also be predicted. In relation to this, Ghargozloo's ionic diffusion oxidation model exists. Therefore, in this paper, an modified oxidation model based on Ghargozloo's oxygen diffusion in uranium dioxide is presented and the volume change of uranium dioxide due to oxidation is predicted.

Keywords: Uranium dioxide, Oxidation, Nuclear fuel, Measurement

A Method for Rapid Determination of ^{99}Tc in Urine Samples

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Technetium-99 (^{99}Tc) is a challenging radionuclide from presents many problems related to safe disposal. The measurement of ^{99}Tc is of particular interest due to its high mobility, and the fact that it is a beta-emitter with a long half-life ($t_{1/2}=2.13\times 10^5$ years) with long-term radiological effects^[1].

As an isotopes of ^{99}Tc , $^{99\text{m}}\text{Tc}$ has been widely applied for medical diagnosis and medical research. It is reported that the $^{99\text{m}}\text{Tc}$ has been used in 80% of diagnostic nuclear medicine procedures and almost 30 million examinations are conducted worldwide using this isotope. Because $^{99\text{m}}\text{Tc}$ has a short half-life of 6 h and decays to ^{99}Tc , monitoring and safe disposal of ^{99}Tc from human urine is very important, and concern is increasing every day as global use of $^{99\text{m}}\text{Tc}$ has increased by more than 4.5×10^{14} Bq per week and is increasing continuously^[2]. However, the current methods for the detection of this radionuclide in such mdium are time consuming and can not satisfy for the low level urine sample analysis.

In this work, a method for rapid determination of ^{99}Tc in urine samples was developed. The sample was firstly pre-treated with $\text{K}_2\text{S}_2\text{O}_8$ to decompose the organic matters combined with ^{99}Tc in $0.5\text{ mol}\cdot\text{L}^{-1}$ HNO_3 medium at 100°C . Then the sample solution was loaded to a TEVA column (2 mL) for ^{99}Tc separation and purification. The target element was finally measured by high resolution inductively coupled plasma mass spectrometry (HR-ICPMS). The developed analytical method was proved to be reliable and can be used to rapid determine low-level ^{99}Tc in urine samples.

Acknowledgements

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Keywords: Technetium-99, Urine sample, Rapid determination

Effect of Hydrogen Peroxide on Electrochemical Behavior of Iron in Perchlorate Solutions

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The effect of hydrogen peroxide on the electrochemical behavior of iron was investigated in perchlorate solutions. Iron showed four distinct behaviors in the perchlorate solutions of pH 3.0. First, the active dissolution regions of Fe with two current waves were observed in the potential range of -0.7 to 0 V (vs. SCE). Second, the stable passivation was found in the potential range between 0 and 0.3 V (vs. SCE). Third, unstable passivation region was observed in the potential range of 0.3 to 1.2 V (vs. SCE). Finally, pitting corrosion was observed at a potential above 1.2 V (vs. SCE). The pH increase stabilized the passivation process of iron, whereas the increase in temperature had a negative influence by enhancing the passivation and pitting behaviors of iron. The presence of hydrogen peroxide at the concentrations below 1.45 mM had an adverse effect on the formation of the passive layer. However, at concentrations above 1.45 mM, hydrogen peroxide affected a beneficial influence on the formation of stable iron oxide layer in the active dissolution region. In addition, regardless of the hydrogen peroxide concentration, the presence of hydrogen peroxide mitigated the pitting corrosion of iron.

Acknowledgements

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Keywords: Hydrogen peroxide, Electrochemical behavior, Iron electrode, Perchlorate

Density Estimation of NaCl-MgCl₂-UCl₃ Melts Using Redlich-Kister Polynomial Equation

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Density of chloride molten salts is an essential physical property in the reactor core design and thermal-hydraulic design simulation, especially in molten salt reactor (MSR) design currently under development in Korea. NaCl-MgCl₂-UCl₃ pseudo-ternary system is one of the various candidate chloride-based salt mixtures because it has relatively-low melting point, very low vapor pressure, high thermal conductivity, etc. However, to the best of our knowledge, the density data of NaCl-MgCl₂-UCl₃ have not yet been measured or published worldwide, and therefore the ballpark figures of the density should be given for the preliminary reactor design. In our present study, the density estimation of NaCl-MgCl₂-UCl₃ based on the pseudo-binary data, i.e., NaCl-MgCl₂, MgCl₂-UCl₃, and NaCl-UCl₃, reported in the literature previously were performed using the Redlich-Kister model. Binary interaction parameter for MgCl₂-UCl₃ was higher than that for NaCl-MgCl₂ and lower than that for NaCl-UCl₃. As an example, calculated density of 0.62 NaCl: 0.18 MgCl₂: 0.20 UCl₃ at 873 K was 2.578 g·cm⁻³. In our further study, the methodology using Redlich-Kister model will be applied to more complex multicomponent systems and to other physical properties such as viscosity, thermal conductivity, surface tension, etc.

Keywords: Density estimation, Molten salt reactor (MSR), NaCl-MgCl₂-UCl₃, Redlich-Kister model

Influence of Fulvic Acid on the Sorption of Se(IV) to Montmorillonite at Acidic Conditions

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The sorption behavior of Se(IV) on montmorillonite clay, a promising buffer and backfill material, was investigated in the presence of aquatic fulvic acid. Selenium-79 is one of the major radioactive nuclides which are long-lived and highly mobile in subsurface environments. Moreover, it is highly toxic even in small amounts, so the selenium quantity in soil and groundwater should be assessed. Although natural organic matters such as humic and fulvic acids are present in the environment, the influence of natural organic matters on Se(IV) migration has not yet been extensively studied.

The batch sorption experiments were performed under oxic conditions. Suwannee River III standard aquatic fulvic acid (International Humic Substances Society) was used to build an organic-rich environment. The N_2 – BET surface area of the montmorillonite (Clay Minerals Society) was $97 \pm 5 \text{ m}^2 \cdot \text{g}^{-1}$. The montmorillonite suspensions with/without fulvic acid were equilibrated with air before adding Na_2SeO_3 . The solid-to-liquid ratio was $5 \text{ g} \cdot \text{L}^{-1}$, the ionic medium was 0.1 M NaCl , fulvic acid concentration was $50 \text{ mg} \cdot \text{L}^{-1}$, and the final pH was 3. The horizontal vial roller was used to prevent the clay from sinking. After 7 days of sorption at room temperature, the suspensions were centrifuged at $10,600 \text{ g}$ for 15 min and filtered through $0.2 \mu\text{m}$ PTFE filters. The colloidal fulvic selenide and free Se(IV) concentrations were entirely measured by inductively coupled plasma–mass spectrometry (ICP-MS).

The sorption results were fitted with Langmuir and Freundlich models. At concentrations lower than $20 \mu\text{M}$, the distribution coefficients (K_d) were $50 \pm 9 \text{ L} \cdot \text{kg}^{-1}$ without fulvic acid, and $36 \pm 5 \text{ L} \cdot \text{kg}^{-1}$ with $50 \text{ mg} \cdot \text{L}^{-1}$ fulvic acid. For the concentrations between 20 and $100 \mu\text{M}$, the K_d values without and with fulvic acid were $16 \pm 7 \text{ L} \cdot \text{kg}^{-1}$ and $10 \pm 1 \text{ L} \cdot \text{kg}^{-1}$, respectively. As a result, it turned out that fulvic acid interferes with the sorption of Se(IV) on montmorillonite in competition with the selenite anion. This indicates that such organic matter may facilitate the migration of selenium in deep geological groundwaters.

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Keywords: Selenium(IV), Fulvic acid, Sorption, Montmorillonite

Current Status on Laboratory Information Management System in KAERI

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According to the article 18 of NSSC notice “Regulations on the delivery of low-and intermediate-level radioactive wastes”, the consignor shall establish and implement the quality assurance program about waste management to ensure conformity with the criteria set forth in the regulations and detailed criteria proposed by the disposal facility operator, including matters related to characterization of the waste concerned. To meet the above requirement, commercially available laboratory information management system, STARLIMS from Abbott Informatics was introduced in the late of 2019 and was customized to our standardized test method in 2020. In that time, Electronic Lab Notebook (ELN), which is an electronic system to create, store, retrieve, and share fully electronic records, was tailored to replace paper lab notebook. Scientific Data Management System (SDMS), which is computer system used to capture, centrally store, catalog, and manage data, was installed. Due to the parsing ability of SDMS, human error like mistake while data entry was reduced by extracting data from measurement sheet and exporting measurement data to designated area of ELN and this feature made work efficiency improved. Afterward, validation of STARLIMS was conducted following the procedure of user acceptance testing including Operational Qualification and Performance Qualification. As a result of these activities, STARLIMS has been officially operated and applied to means to manage test results since 2021. In 2021, for user-friendly environment, updating STARLIMS was also conducted by applying SDMS to import data from other radiometric measurements including gas proportional counter (GPC), liquid scintillation counter (LSC), and low-energy Ge detector (LEGe) besides HPGe detector for gamma measurement.

From implementation to operation, it is confirmed that STARLIMS has been providing reliable and stable platforms to manage laboratory information regarding measurement records and playing a significant role in tool to meet the quality assurance required.

Keywords: Laboratory information management system, Quality assurance, User acceptance testing, Electronic lab notebook, Scientific data management system

7분과

방사화학(Poster)

Structural Variations of Spontaneously Formed Crystalline Eu(III)-Aliphatic Dicarboxylate Compounds Studied by Powder XRD and Time-resolved luminescence Spectroscopy

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Complexation of actinides and lanthanides with carboxylic organic ligands is known to facilitate migration of radionuclides from deep geological disposal systems of spent nuclear fuel. In order to examine the ligand-dependent structures of trivalent actinides and lanthanides, a series of Eu(III)-aliphatic dicarboxylate compounds, $\text{Eu}_2(\text{oxalate})_3(\text{H}_2\text{O})_6$, $\text{Eu}_2(\text{malonate})_3(\text{H}_2\text{O})_6$, and $\text{Eu}_2(\text{succinate})_3(\text{H}_2\text{O})_2$, were synthesized and characterized by using X-ray crystallography and time-resolved laser fluorescence spectroscopy. Powder X-ray diffraction results captured the transition of the coordination modes of aliphatic dicarboxylate ligands from side-on to end-on binding as the carbon chain length increases. This transition is illustrated in malonate bindings involving a combination of side-on and end-on modes. Strongly enhanced luminescence of these solid compounds, especially on the hypersensitive peak, indicates a low site symmetry of these solid compounds. Luminescence lifetimes of the compounds were measured to be increased, which is ascribed to the displacement of water molecules in the inner-sphere of Eu center upon bindings of the organic ligands. The numbers of remaining bound water molecules estimated from the increased luminescence lifetimes were in good agreement with crystal structures. The excitation-emission matrix spectra of these crystalline polymers suggest that oxalate ligands promote the sensitized luminescence of Eu(III), especially in the UV region. In the case of malonate and succinate ligands, charge transfer occurs in the opposite direction from Eu(III) to the ligands under UV excitation, resulting in weaker luminescence.

Keywords: Lanthanide, X-ray crystallography, Coordination chemistry, Time-resolved luminescence spectroscopy, Excitation-emission matrix

Deduction of Appropriate Measurement Time for Gamma Radionuclide Analysis Through the Minimum Detectable Activity of Unknown Sample

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Appropriateness of the minimum detectable activity in the analysis of gamma radionuclides is very important. This is reason determine the time factor among the conditions of the analysis when it is rationally determined has the advantage that radioactivity analysis can be performed accurately and quickly.

In this study, 100 mL of an unknown sample was diluted in Marinelli Beaker 1L to obtain, review data on gamma radiation analysis results and minimum detectable activity for each measurement time. The measurement was used High Purity Germanium detector, target nuclides are Co-57, Co-58, Y-88 and Cs-137. Since the radioactivity analysis sample will be expected to be the waste subject to self-disposal or less during the radioactive waste classification, the minimum detectable activity standard was set based on the detection of less than the permissible activity for self-disposal for each nuclide.

The measurement methods were measured by classifying it into seven categories: 1000 seconds, 3600 seconds, 10000 seconds, 30000 seconds, 80000 seconds, 100000 seconds, and 150000 seconds. The radioactivity from this measurement are Co-57 $2.89 \text{ Bq} \cdot \text{g}^{-1}$, Co-58 $0.19 \text{ Bq} \cdot \text{g}^{-1}$, Y-88 $0.20 \text{ Bq} \cdot \text{g}^{-1}$, Cs-137 $0.15 \text{ Bq} \cdot \text{g}^{-1}$, the measurement results under all conditions were similar. On the other hand, the minimum detectable activity showed values above the allowable activity for self-disposal in not but Co-58 at 1000 and 3600 seconds. Only after taking the measurement time of 10000 seconds, the result was derived Co-57 $0.0095 \text{ Bq} \cdot \text{g}^{-1}$, Co-58 $0.0068 \text{ Bq} \cdot \text{g}^{-1}$, Y-88 $0.0052 \text{ Bq} \cdot \text{g}^{-1}$, Cs-137 $0.0062 \text{ Bq} \cdot \text{g}^{-1}$, which was confirmed to less than the allowable activity for self-disposal by nuclide.

Reasonably determining the measurement time in gamma radionuclide analysis is a very important issue in terms of economy of time and accuracy of measurement. Although this study cannot be said to be able to determine a reasonable measurement time for all gamma radionuclide analysis, it is hoped that research on various samples will be made to contribute to the efficient measurement of gamma radioactivity.

Keywords: Gamma radioactivity analysis, Minimum detectable activity, Measurement time, High purity germanium detector

Spectroscopic Studies on Chemical Equilibria of a Catechol Ligand: Acid Dissociation, U(VI) Complexation and Effect of Ionic Strength

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To predict the long-term behaviors of actinides in aqueous environments, complexation behaviors of actinides should be understood. Various organic ligands of chelating aromatic structure appearing in humic substances are known to form stable aqueous complexes. In this study, a benzene diol (or catechol) derivative, i.e., 4-nitrocatechol (nCA) is selected and its chemical equilibria including acid dissociation and complexation with U(VI) ion were examined using spectroscopic methods. In addition, the effect of ionic strength (I_s) on those equilibria was evaluated by adjusting the level of NaClO_4 in aqueous solutions. First, the experiments to determine the acid dissociation constant (pK_a) of nCA were carried out in aqueous solutions with different ionic strengths from 0.01–2.0 M. The acid dissociation constants of nCA (pK_{a1}) were measured to 6.73 ± 0.07 , 6.69 ± 0.03 , 6.38 ± 0.03 , 6.09 ± 0.12 , and 6.04 ± 0.07 at $I_s = 0.01, 0.1, 0.5, 1.0$, and 2.0 , respectively. These results were confirmed through the UV-Vis absorption spectral data analysis using the HypSpec program. As the pK_{a1} decreases as the ionic strength increases, except for $I_s = 2.0$, these data were further analyzed with SIT (Specific ion Interaction Theory). Typically, as the solution becomes basic, a red shift is shown in the absorption spectrum. This effect can be understood from the intramolecular charge transfer (ICT) occurring in the deprotonated structures of nCA. At higher pH similar trends were also observed for measurement of pK_{a2} . However, the determination of pK_{a2} is found not to be straightforward since a dimer formation equilibrium of nCA was observed as the ionic strength increased. This phenomenon will be discussed in detail with other supporting experimental results. Second of all, the complexation between the U(VI) and nCA in aqueous solutions was also examined. It was shown that nCA can easily form complexes with U(VI) ions at a wide range of pH via the deprotonation of their hydroxyl groups. U(VI)-nCA complexation will be further characterized by UV-Vis spectroscopy, IR and NMR by varying the solution ionic strength. The metal-ligand binding stoichiometry will be confirmed, for example, through the Job's method. Finally, the acid dissociations constant and stability constants that were determined in this study will be used to provide species diagrams of aqueous U(VI)-nCA systems at a wide range of pH considering the effect of solution ionic strengths.

Keywords: 4-Nitrocatechol, Acid dissociation constant (pK_a), Hexavalent uranium, Uranium complexation, Stability constant ($\log\beta$), Coordination chemistry, Spectroscopic method

A Rapid and Quantitative Sequential Separation Method for H-3 and C-14 Isotopes With Distillation Apparatus in Environmental Samples

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This study presents a rapid and quantitative sequential separation method for H-3 and C-14 isotopes with distillation apparatus in environmental samples released from nuclear facilities. After adding 200 mg of granulated potassium permanganate and 500 mg of sodium hydroxide in 100 mL of sample solution, the sample solution was heated until approximately 10 mL of distillate, and the distillate fraction was removed. The sample solution was heated again until a minimum 10 mL of additional distillate was collected. 10 mL of distillate was transferred to the LSC vial and the measurement sample for H-3 was made by adding 10 mL of Ultima Gold LLT to the LSC vial. After adding 2.5 g of potassium persulfate, 2 mL of 1M silver nitrate and 15 mL of concentrated nitric acid to the remained sample solution, the sample solution was heated for 90 minutes and C-14 isotopes were adsorbed into 10 mL of Carbo-Sorb solution in glass vial. The measurement sample for C-14 was made by adding 10 mL of Permafluor to the C-14 fraction in glass vial. The purified H-3 and C-14 samples were measured by the liquid scintillation counter after quenching correction. The average recoveries of H-3 and C-14 with CRM were measured to be 96% and 85%, respectively. The sequential separation method for H-3 and C-14 investigated in this study was applied to activated charcoal filter produced from nuclear power plants after validating the reliability by result of proficiency test (KOLAS-KRISS, PT-2021-51).

Keywords: Sequential separation, H-3, C-14, Distillation method, Activated charcoal filter, Proficiency test

A Simulation Study to Evaluate the Amount of C-14 Production According to Location in Neutron-irradiated Graphite

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Graphite is widely used as a reflector or moderator in nuclear reactors, and since it is exposed to high flux neutron irradiation, long-lived radioactive isotopes such as C-14 are formed. Therefore, quantitatively prediction of production amount is a very essential task for reliable radioactive waste management. In this study, considering nuclear reactions such as (n, γ) reaction by thermal neutrons and elastic scattering, various characteristics such as the rate of formation of C-14 and energy distribution of thermal neutrons according to the penetration depth from the graphite surface were numerically analyzed. The evaluation was carried out in consideration of the average energy of neutrons and reaction/collision cross-sectional area at given energy, and a comparative study was also performed when the thermal neutrons were in thermal equilibrium and when they were not. The numerically evaluated results were compared qualitatively with the experimental study, and methods to further increase the accuracy were also discussed.

Keywords: Irradiated graphite, Carbon isotopes, Numerical analysis, Neutron absorption, Maxwell-boltzmann distribution

Temperature-Dependent Adsorption of U(VI) Onto γ -Alumina

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An elevated temperature is expected at the deep geological repository (DGR) due to the decay heat from spent nuclear fuel and the positive geothermal gradient. The resulting elevated temperature would change the aqueous speciation and surface complexation of uranium, which is the major component in spent nuclear fuel. Since sorption reactions of uranium species on natural minerals determine the extent of uranium retardation, in this work the temperature-dependent adsorption of hexavalent uranium, U(VI), was studied by choosing alumina as the basic component mineral for complex aluminosilicates.

Time-resolved laser fluorescence spectroscopy (TRLFS) was used to assess the dissolved and adsorbed U(VI) species on γ -Alumina in the pH range of 6.5–9.0 at temperatures of 25 to 70°C. Initial concentrations of U(VI), carbonate and calcium were 89 μ M, 25 mM, and 3.0 mM, respectively. The parallel factor analysis (PARAFAC) was used for chemical speciation by spectrum deconvolution. In addition, a separate solution system with higher U(VI) concentrations (0.1 mM, 1.0 mM) and carbonate concentration of 25 mM was studied with attenuated total reflectance-Fourier transform infrared (ATR-FTIR) spectroscopy for adsorbed species at 25°C. The electrophoretic mobility measurements were also conducted at 25°C to assess the coordination mechanism of adsorbed species at 25°C.

The uranyl hydrolysis species and uranyl tricarbonate species coexist in solution at 25°C. At the same temperature, both species were found to be adsorbed. ATR-FTIR could confirm the adsorption of uranyl tricarbonate species at 25°C, and the electrophoretic mobility measurements suggested that the reaction mechanism is an inner-sphere coordination. However, in comparison with aqueous speciation at 25°C, at elevated temperatures the available pH range of uranyl tricarbonate species was narrow and that for uranyl hydrolysis species was wider. It was evident that two hydrolysis species are adsorbed at elevated temperatures, but no tricarbonate species. The enhanced U(VI) adsorption was observed with temperatures. This could result from the transition of dominance from the concurrent adsorption of uranyl hydrolysis species and uranyl tricarbonate species to two hydrolysis species. It was seen that the trend of enthalpy of adsorption was endothermic. Combining the present results with temperature-dependent adsorption studies on silica and aluminosilicates, a reliable SCM for the subsurface system can be proposed to explain U(VI) migration.

Acknowledgements

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Keywords: Uranium, Alumina, Elevated temperatures, Adsorption, Uranyl hydrolysis species, Uranyl tricarbonate species

Quantification of Fission Products in Molten Salts Using Laser-induced Breakdown Spectroscopy With Machine Learning Technology

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Molten salt reactors and pyroprocessing are widely considered for various nuclear applications. The main challenges for monitoring these systems are high temperature and strong radiation. Two harsh environments make the monitoring system needs to measure nuclides at a long distance with sufficient resolution for discriminating many different elements simultaneously. Among available methodologies, laser-induced breakdown spectroscopy (LIBS) has been the most studied. The LIBS method can provide the required stand-off and desired multi-elemental measurable ability. However, the change of the level for molten salts induces uncertainty in measuring the concentration of the nuclides for LIBS analysis. The spectra could change by focusing points due to the different laser fluence and plasma shape.

In this study, to prepare for such uncertainties, we evaluated a LIBS monitoring system with machine learning technology. While the machine learning technology cannot use academic knowledge of the atomic spectrum, this technique finds the new variable as a vector from any data including the noise, target spectrum, standard deviation, etc. Herein, the partial least squares (PLS) and artificial neural network (ANN) were studied because these methods represent linear and nonlinear machine learning methods respectively. The Sr (580–7200 ppm) and Mo (480–4700 ppm) as fission products were investigated for constructing the prediction model. For acquiring the data, the experiments were conducted at 550°C in LiCl-KCl using a glassy carbon crucible. The LIBS technique was used for accumulating spectra data. In these works, we successfully obtained a reasonable prediction model and compared each other. The high linearities of the prediction model were recorded. The R^2 values are over 0.98. In addition, the root means square of the calibration and cross-validation were used for evaluating the prediction model quantitatively.

Acknowledgements

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Keywords: Laser-induced breakdown spectroscopy, Pyroprocessing, Molten salt reactor, Machine learning, Partial least squares, Artificial neural network

Chemical Characteristics for the CeO₂ Samples Containing Mo Epsilon Particle

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To evaluate the chemical properties of the epsilon particle present as a precipitate in the high-level radioactive waste, we performed the experiments for the samples fabricated by CeO₂ containing 1, 3, and 5wt%Mo, where CeO₂ is used as the simulated high-level radioactive waste to replace the real one, using X-ray diffraction (XRD), X-ray absorption spectroscopy (XAS), X-ray photoelectron spectroscopy (XPS), and scanning electron microscopy-energy dispersive X-ray spectroscopy (SEM-EDS). Moreover, to evaluate the chemical behavior of Mo epsilon particle in CeO₂ at high temperature, the manufactured CeO₂-1wt%Mo, CeO₂-3wt%Mo, and CeO₂-5wt%Mo samples were heated at 100, 300, 500, 700 and 900°C for 7 h in a tube furnace under Ar atmosphere. In this study, the results of comprehensive analysis including the crystal structure, chemical state, and elemental distribution will be presented to verify the chemical properties for CeO₂ samples containing Mo epsilon particle, depending on the Mo content and heating temperature.

Keywords: High-level radioactive waste, Epsilon particle

Automated Non-Volatile Radionuclide Separation System for Prompt Analysis of LILWs

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According to Article 4 and 5 of the Nuclear Safety and Security Commission (NSSC) Notice No. 2020-6, radioactive waste packages should be classified by radioactive levels, and finally permanently shipped to underground or surface disposal facilities. The level of the radioactive waste package is determined based on the concentrations of the radionuclides suggested in Article 8 of NSSC Notice No. 2021-26. Since most of the radionuclides in radioactive wastes are beta nuclides, chemical separation and quantification of the target nuclides are essential.

Conventional methods to classify chemically non-volatile radionuclides such as Tc-99, Sr-90, Nb-94, Fe-55 take a lot of time (about 5 days) and have low efficiency. An automated non-volatile nuclide analysis system based on the continuous chemical separation method of radionuclides has been developed to compensate for this disadvantages of the conventional method in this study.

The features of the automated non-volatile nuclide separation system are as follows.

First, the amount of secondary waste generated during the chemical separation process is very small.

That is, by adopting an open-bed resin column method instead of a closed-bed resin column method, additional fittings and connector are unnecessary during the chemical separation. In addition, because the peristaltic pump is supplied for the sample and solution respectively, it is great effective to prevent cross-contamination between radioactive samples and the acid stock solution for analysis.

Second, the factors that may affect results, such as solution amount, operating time and flow rate, are almost constant.

By mechanically controlling the flow rate precisely, the operating time and additional factors required during the separation process can be adjusted and predicted in advance, and the uncertainty of the chemical separation process can be significantly reduced.

Finally, it is highly usable not only in the continuous separation process but also in the individual separation process. It can be applied to the individual separation process because the user can set the individual sequence using the program.

As a result of the performance evaluation of the automation system, recovery rates of about 80–90% and reproducibility within 5% were secured for all of the radionuclides. Furthermore, it was confirmed that the actual work time was reduced by more than 50% compared to the previous manual method. (It was confirmed that the operation time required during the separation process was reduced from 6 days to 3 days.)

Based on these results, the automation system is expected to improve the safety of workers in radiation exposure, reduce human error, and improve data reliability.

Acknowledgements

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Keywords: Non-volatile radionuclide, Automation, Radioactive wastes, nuclide separation

Effects of Tin, Niobium, and Iron Elements in Zirconium-based Samples on Lattice Thermal Expansion

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The lattice thermal expansion of zirconium-based samples containing tin, niobium, and iron elements at a temperature range of 30–870°C with intervals of 40°C was studied by *in situ* high-temperature X-ray diffraction (HT-XRD). The *a*- and *c*-axes lattice constants of the hexagonal Zr crystal structure for the zirconium-based samples were calculated by Pawley refinement using the *in situ* HT-XRD spectra. The *a*-axis lattice parameters for the zirconium-based samples with tin element overall decreased, whereas those for the samples containing niobium or iron elements are not declined, as compared to those for a pure zirconium sample. It suggests that the lattice thermal expansion along the *a*-axis direction of the hexagonal Zr crystal structure for zirconium-based samples was suppressed by the tin element. This effect is the greatest when the content of tin element added in zirconium-based sample is 3wt%. On the other hand, the *c*-axis lattice parameters for all the zirconium-based samples overall increase as compared to the pure zirconium, indicating no suppression effect by tin, niobium, and iron elements, in contrast to the *a*-axis lattice constants.

Keywords: Zirconium, Minor alloying element, Cladding, High-temperature x-ray diffraction, Lattice parameter

Voltammetric Investigation of U and Pu in Perchloric Acid Media Using Gold Working Electrode

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Elucidating the redox behavior of actinide elements in aqueous solution is important for the safety assessments of nuclear waste disposal. Despite ongoing endeavors for decades, some points of uranium and plutonium redox mechanism are ambiguous and unclear. In this study, the electrochemical redox behaviors of U(VI) and Pu(III and VI) ions in perchloric acid media were investigated by using a gold (Au) working electrode via cyclic voltammetry (CV) and cyclic square wave voltammetry (CSWV) with the temperature control (10–55°C). The cyclic voltammograms of U(V/VI), Pu(III/IV) and Pu(V/VI) redox couple were transformed to semi-integral form to calculate the diffusion coefficient and formal potential in the electrochemical quasi-reversibility prevailed system. The CSWV was additionally used for a more precise interpretation of the redox mechanism. From the investigation of the redox chemistry of U(VI) ions, a clear U(V/VI) redox peak and one unidentified oxidation peak appear around pH 2. With the temperature control and CSWV, the relevance of the oxidation peak and U(IV) was confirmed. In the case of voltammetry of Pu(VI) solution, Pu(V/VI) redox peak and an additional reduction peak appear. The redox behavior responsible for this additional reduction peak are also examined. The cyclic voltammograms of Pu(III) solution show a clear reversible redox reaction of Pu(III/IV) couple. With the temperature control, using the change of formal potential at ionic strength 1 M (ClO_4^-), thermodynamic parameters of conditional molar enthalpy and entropy change were evaluated in this system.

Keywords: Uranium, Plutonium, Cyclic voltammetry, Cyclic square wave voltammetry, Thermodynamics

Evaluation of the Corrosion Resistance for Alloy 600 and 617 Against KCl Molten Salt

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A molten salt reactor (MSR) has considerably attracted attention due to its several advantages for the safety and efficiency over the light water reactors. Because the structural material in MSR is contacted with high-temperature liquid fuel during long-term, the excellent material for corrosion resistance is required to be applied in MSR. In this study, we evaluated the corrosion resistance for alloy 600 and 617, which are the nickel-based materials, in KCl molten salt at 800°C for 100 h under Ar atmosphere containing less than 1 ppm of moisture and oxygen. After the corrosion experiments of alloy 600 and 617, the amount of the weight loss for them caused by the KCl molten salt were determined. In addition, the variation in the crystal structure, surface morphology, and elemental distribution was examined using X-ray diffraction and scanning electron microscopy equipped with energy dispersive X-ray spectroscopy.

Keywords: MSR, Corrosion, Ni alloy

Synthesis and Characterization of New Copper Based 2D MOFs With Subsequent Application for I₂ Trapping

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Efficient capture and storage of radioactive iodine (consisting of two isotopes: ¹²⁹I and ¹³¹I), produced or released from nuclear activities, are of paramount importance for sustainable development of nuclear energy due to their volatility and long half-life. Therefore, it is very important to develop new adsorbents for efficient utilization of radioactive iodine from nuclear waste. Various methods and materials are used for I₂ capturing and removing, including MOFs due to their high porosity and fast adsorption kinetics, which are rightfully considered effective sorbents for removing I₂. Metal-organic frameworks (MOFs) are porous crystalline materials which have diverse pore geometry and unique physicochemical properties, have attracted enormous attention for use in gas storage, separation and catalysis. The ability of MOFs to adsorb volatile products at room temperature can significantly improve the cost-effectiveness of the utilization process.

This work describes the synthesis and characterization of three new metal-organic frameworks based on pyrazine (pyz), 4,4'-bipyridine (bpy), 1,2-bis(4-pyridyl)-ethane (bpe) and copper (II) hexafluoroantimonate, as potential adsorbents for I₂ capture. All of these three MOFs exhibit a two-dimensional (2D) crystal structure consisting from infinity non-crossing linear chains. The crystal structure of [Cu(pyz)₂(ZrF₆)₂(H₂O)₂], [Cu(bpy)₄(H₂O)₂ZrF₆] and [Cu(bpe)₄(H₂O)₂ZrF₆] were characterized using powder X-ray diffraction (PXRD), single crystal X-ray diffraction (SC-XRD). Comparative characteristics of synthesized MOFs, including Fourier-transform infrared spectroscopy (FTIR) and thermogravimetric analysis (TGA) were also performed. The I₂ sorption experiments were examined by UV-vis spectroscopy.

Keywords: Iodine capture, Metal-organic frameworks, SC-XRD

Synthesis of New Anderson-Type Polyoxometalate With Titanium as a Heteroatom, and Structural Characterization

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Polyoxometalates (POMs) are nanoclusters composed of transition metals with high oxidation states. Owing to their redox properties and structural diversity, POMs have been applied to broad fields, such as catalysis, materials, and medicine. Among various fields of application, POMs play an important role in radiochemistry. POMs can form complexes with tri- and tetravalent lanthanides and actinides (radioactive elements), which may be good sequestrators or agents for separating nuclear wastes.

Among the most prominent POM structures, Anderson-type POMs with a general formula of $[H_y(XO_6)M_6O_{18}]^{n-}$ ($y=0-6$, $n=2-8$, M =addenda atom, X =heteroatom) represent one of the basic topological structures of the POM family. An important feature of Anderson type POMs is incorporating a large number of various heteroatoms with different size and oxidation states, which can lead to tune chemical properties. Interestingly, no example of Anderson type POMs with early transition metal ions in the heteroatom site has been reported to date. Herein, we discovered that the Anderson POM $Na_2K_6Ti_{0.92}W_{6.08}O_{24} \cdot 12H_2O$, which consists of pure inorganic framework built from a central Ti core supported by six WO_6 inorganic scaffold, and the crystal structure was confirmed and refined using single-crystal X-ray diffraction (SC-XRD). In addition, structural characterizations, including, Fourier-transform infrared spectroscopy (FT-IR), X-ray photoelectron spectroscopy (XPS), and Inductively coupled plasma-optical emission spectroscopy (ICP-OES) were performed.

Keywords: Anderson-type polyoxometalate, Titanium, Synthesis, Structural characterization

Comparison of the Effects of Solid Solution Elements and Epsilon Particles on the Structure and Properties of Nuclear Fuel

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Spent nuclear fuel is a very complex material because various elements such as fission products, transuranium elements and activation products are produced from initial fresh UO_2 fuel after irradiation. These elements exist in UO_2 with various forms and can change the structure and of physicochemical properties of UO_2 . These changes could provide the surface activation site that could enhance chemical reactions and corrosion processes, and would significantly affect the storage environment for long-term disposal of spent nuclear fuel. Therefore, it can be important to understand the characteristics of spent nuclear fuel to design reliable and safe geological repositories. However, it is too hard to study the characteristics of spent nuclear fuel, because it is a very complex material by itself and not easy to handle due to its radioactivity, and it is also difficult to independently understand the effects of each element. Therefore, a simulated spent nuclear fuel containing an element that forms a solid solution and epsilon particle was manufactured to understand the change in characteristics of each element. Most of the elements that form solid solutions are lanthanides or actinides and can change the structure of the UO_2 lattice itself. The epsilon particles exist as metals at the grain boundaries of UO_2 . In this study, structural changes were measured using XRD, SEM, and Raman spectroscopy, and physical and chemical properties were also identified by measuring electrical conductivity and electrochemical properties. The results were summarized, and the effects of solid solution elements and epsilon particles on the structure and properties of UO_2 matrix were compared and discussed.

Acknowledgements

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Keywords: Uranium dioxide, ϵ -particle, Solid solution, Surface structure, Chemical property

Preliminary Estimation of Thermophysical Properties for Ternary Molten Chloride System NaCl-MgCl₂-UCl₃

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A molten salt reactor (MSR) that uses molten salt mixtures as nuclear liquid fuel has recently received much attention due to its inherent safety. Various fluoride and chloride salt mixtures are considered as fluid fuel for MSRs. Among those, NaCl-MgCl₂-UCl₃ system is the one of the most promising candidates for molten salt fast reactor. The comprehensive information on thermo-physical properties such as density, viscosity, heat capacity and thermal conductivity are fundamental to MSR design development, but experimental data for NaCl-MgCl₂-UCl₃ system are unknown to the best of our knowledge. In this study, we estimated the thermophysical properties of NaCl-MgCl₂-UCl₃ system. The properties were calculated by mole fraction additive method using reliable experimental data from pure salt system. Other methods, such as rule of additivity of molar volume for density, modified Dulong-Petit method for heat capacity, and Rao-Turnbull prediction and Ignatieve-Khokolve correlation for thermal conductivity, have also been applied. Estimated values for the properties were compared with each other as well as available binary experimental data.

Keywords: Molten salt reactor, Thermophysical properties, Density, Viscosity, Heat capacity, Thermal conductivity

Development of an Automated Method for Determination of Chelating Agents in Radioactive Wastes

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Chelating agents, such as EDTA, NTA, and citric acid, can form radionuclide-chelate complexes that may enhance the migration of radionuclides from disposal sites. Therefore, the contents of these chelators in radioactive wastes are determined for the secured disposal.

In this study, a rapid automated method using ion chromatography (IC) is described for analysis of chelating agents. The method enables direct use of a sample solution for the measurement unlike colorimetric or enzymatic methods. In these cases, lots of chemicals are consumed to convert the chelating agents to be UV-Vis-active compounds during the pretreatment process and finally, absorbance or transmittance are measured at specific wavelength using UV-Vis spectrophotometer. In particular, the enzymatic method for determination of citric acid, because of its strong dependence on experimental conditions including sample types and chemical reagents may produce inconsistent results. The automated system using IC allows a laboratory to directly measure the amount of citric acid while reducing total analysis time and increasing efficiency. In addition, this method is capable of detecting a trace amount of citric acid, thereby lowering LOD and LOQ values.

Keywords: Chelating agents, Automated method, Ion chromatography, Colorimetric method, Enzymatic method, UV-Vis spectrophotometer

Correction Method for High-Temperature Low-Viscosity Measurement System Using Narrow-Gap Rotational Viscometer

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Measurement of the physical properties of high-temperature molten salts is important for the efficient design and operation of molten salt reactors (MSR) in which the reactor coolant and nuclear fuel are in a homogeneous liquid state. Although some crucial physical properties such as viscosity, thermal conductivity, density, etc., have been drawing much attention, relative data, especially for molten chloride salts, are scarce. Thus, it is urgent to prepare the viscosity data as one of the key transport properties in thermal hydraulics analysis. However, it is not an easy task to measure the molten salt viscosity with high accuracy due to end effect, a small gap between the chamber and spindle, thermal expansion of the chamber and spindle at high temperatures in a rotational viscometer. Additionally, molten salt temperatures inside furnace are not uniform due to the large temperature gradient inside the chamber, and therefore the assumption of laminar condition can be violated.

In this study, geometric factors, which can be a major interference in the torque measurement, were considered for the accurate determination of the viscosity. We established a high-temperature molten salt viscosity measurement system with Brookfield rotational viscometer. KNO₃ molten salt was used as a model substance at a temperature range of 650–773 K. In-house designed spindles and chambers were made of corrosion-resistant alumina. Thermal expansion has a significant influence on the size and shape of the chamber and spindle. The effect of thermal expansion on the conventional correction method was examined with temperature variation and distribution. Gap size variation was also investigated in order to improve the accuracy.

Keywords: Viscosity, Correction, Rotational viscometer, Molten salt

Outstanding Resistance of Gd-doped UO_2 Against Surface Oxidation

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Accurate understanding of structural integrity and chemical reactivity of UO_2 disposed in deep underground sites is of importance. Owing to the specific condition of the site location, UO_2 may have substantially different properties from the conventional prediction. In this study, we demonstrate that the oxidation resistivity of UO_2 is considerably modified by gadolinium (Gd), which is the element of neutron absorber and a byproduct of nuclear decay of radioactive U-235. Using density functional theory calculations, we investigate how the oxidation mechanism of UO_2 changes with Gd incorporation in U lattice. Our study indicates that Gd remarkably enhances the thermodynamic stability of pristine UO_2 against surface oxidation via three underlying mechanisms: (i) weakens the chemical bonding of adsorbed oxygen atom (O) with U, (ii) reduces active sites (U) for oxygen adsorption, and (iii) suppresses the subsurface diffusion of adsorbed O delaying the growth of the oxide layers on the UO_2 . Electronic and lattice structure analyses for Gd-doped UO_2 indicate that amount of charge transfer from U to O is critically reduced and the lattice of the UO_2 surface is contracted. Our results provide useful information for understanding long-term stability and improving the structural integrity of UO_2 through the chemical doping process.

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Keywords: Density functional theory, Surface oxidation, Gd-doped UO_2 , Subsurface diffusion, Oxide-layer growth

A Study on Molten Salt Purification for Multi-purpose Molten Salt Experiments

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Molten salt used in the multipurpose molten salt experiment must be of high purity. Depending on the purpose of the experiment, only the base component of the molten salt be used, or a component simulating a nuclear fission product be added to the base component and used. In all cases, an increase in the concentration of impurities such as oxygen and moisture may lead to an erroneous interpretation when analyzing the experimental results. Therefore, molten salt should be purified before use. In this study, the purification of molten salt is described for multi-purpose molten salt experiments. The salt mixture is selected as $\text{MgCl}_2\text{-NaCl}$ and is quantified at a mixing ratio of 43mol%:57mol%. The salt mixture is treated in a glove box environment because of must minimize the reaction of adsorbing oxygen and moisture when the salt mixture is exposed to the atmosphere. MgCl_2 is more likely to contain water than NaCl , the purification of the NaCl-MgCl_2 mixture is established according to the purification process for removing water from MgCl_2 . A process for purifying the salt mixture briefly consists as follows: drying moisture, melting salts, purification, removing HCl , and stabilization. Through the process be able to obtain high-purity molten salt and more accurate experiment results.

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Keywords: Molten salt reactor, Molten salt, Liquid fuel, Thermal convection loop

Optimal Operation Method of the Thermal Convection Loop for Multi-purpose Molten Salt Experiments

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At high temperatures, molten salt has heat transfer properties like water. Molten salt has the characteristics of a strong natural circulation tendency, large heat capacity, and low thermal conductivity. Unlike sodium, molten salt does not react explosively exothermically with air. However, molten salt has a strong tendency to corrode materials, and its properties are easily changed by a sensitive reaction to oxygen and moisture. Therefore, it is necessary to study material corrosion properties and chemical control methods for nuclear fuel salts, which are eutectic mixtures. In this study, the optimal operation method of the thermal convection loop is established to perform the experiments on the molten salt. The process describes briefly as follows. The operation step consists of preparation, purification, transportation, and operation. In the preparation, the step checks the entire structure and equipment (TC, blower, vacuum pump, etc.). And melt the salt mixture at a high temperature (670°C) slowly in the purification step. Before injecting the molten salt, the surface temperature of the entire loop must retain temperature (about 500°C) constantly. Completely melted molten salt in the melting pot is flow along the pipe of the thermal convection loop in the transportation step. Lastly, the convection of molten salt goes to keep by the temperature difference. The thermal convection loop can be utilized for various experiments such as corrosion tests, component analyses, chemistry control, etc.

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Keywords: Molten salt reactor, Molten salt, Liquid fuel, Thermal convection loop

Determination of Activity Concentration of ^{63}Ni and ^{59}Ni Using Liquid Scintillation Counter According to $^{63}\text{Ni}/^{59}\text{Ni}$ Ratio

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Radioactive wastes that are generated as a result of operating NPPs, contain ^{63}Ni and ^{59}Ni that should be analyzed in accordance with the notice of Nuclear Safety and Security Commission (NSSC) for the acceptance of Korea Radioactive Waste Agency (KORAD). Analyzing ^{63}Ni and ^{59}Ni has few challenges to determine activities of each nuclide in radioactive waste sample that contains both nuclides. As is well known, ^{63}Ni can be analyzed by liquid scintillation counter (LSC) detecting its emitted beta rays, however, beta rays emitted from ^{59}Ni are overlapped on the spectrum. Therefore, to discriminate those two nuclides, spectrum channel should be divided according to its dedicating part of the spectrum. For instance, ^{59}Ni contribute to spectrum channel 30–250, on the other hand, ^{63}Ni contributes to spectrum channel 30–450. In other word, ^{63}Ni solely can be analyzed on the channel from 260 to 450. To analyze both ^{63}Ni and ^{59}Ni using this channel division method, detection efficiency must be measured in advance; efficiency of ^{63}Ni and ^{59}Ni at ch. 30–250, and efficiency of ^{63}Ni at ch. 260–450, then the activity can be calculated using the corresponding efficiency.

In this study, for verifying the feasibility of channel division method, 5 simulated samples were prepared with different ratio of $^{63}\text{Ni}/^{59}\text{Ni}$. The ratio varies as 1, 2, 10, 20 and 100 spiking standard source of ^{63}Ni and ^{59}Ni . Each sample was mixed with scintillation cocktail and detected for 90 minutes by LSC (300SL, Hidex) after the stabilization of solutions. As a result, calculated ^{63}Ni activities for all sample were averaged as 97% of spiked activity. However, calculated ^{59}Ni activity were 101%, 103%, 128%, 140%, 260%, respectively. The result indicates that ^{59}Ni cannot be discriminated by channel division method when it exists in the sample with high ^{63}Ni over 10 times then ^{59}Ni such as radioactive waste sample. However, the results also show that the channel division method for analyzing ^{63}Ni activity was successful verifying it can determine the activity of ^{63}Ni regardless of the affect of ^{59}Ni on the spectrum.

Keywords: Ni-63, Ni-59, LSC, Channel division, Radioactive waste

Investigation of Thermohydrodynamic Effect on the Rotating Cylinder Electrode in High Temperature Molten Salt System

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Electroanalytical study for the rotating cylinder electrode in molten LiCl-KCl eutectic salt (58–42mol%) containing MgCl₂ (0.1wt%) at 600°C is conducted. The researches of rotating cylinder electrode have been widely conducted for the century. The advantage of the electrode is that it can mitigate the unintended natural convection by providing a controlled diffusion boundary layer thickness. However, the experimental data for the high temperature molten salts is barely existed. The study adopts the electrochemical techniques such as cyclic voltammetry for the static cell and linear sweep voltammetry for the dynamic cell to calculate the diffusion coefficient. The peak current density and limiting current density are measured according to the scan rate. In order to evaluate the mass transfer under hydrodynamic flow condition, the revolution speeds of cylindrical electrode are varied from 10 rpm to 500 rpm which are corresponded to the Reynolds number of 4 and 185 respectively. The flow regime covers from the laminar to semi-turbulent regime (transient) as the critical Reynolds number Re_{crit} is 200. The limiting current density shows a linear trend with the revolution speed and agrees well with the existing mass transfer correlations. For the extended flow regime, a new mass transfer correlation is suggested as the relation of non-dimensional numbers ($Sh = aRe^bSc^c$) based on the dimensionless analysis.

Keywords: Electrochemistry, Electroanalytical technique, Rotating cylinder electrode, Diffusion coefficient, Pyroprocessing

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