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KOREAN RADIOACTIVE WASTE SOCIETY

2021
춘계학술논문요약집

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2021. 6. 2(수) ~ 4(금)

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Zoom

| 오프라인 | 동국제강연수원 후인원

Abstracts of Proceedings of the
Korean Radioactive Waste Society
SPRING 2021

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1분과

핵주기정책· 규제 및 비확산(Oral)

DPRK Delivery Systems in Association With Nuclear Detonations

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Since 2019 the DPRK has continuously developed and demonstrated her impressive weapon delivery systems such as ballistic as well as cruise missiles, multiple rock launch systems, and SLBM's. Most of these development products were demonstrated throughout two big parades in 2020 October and 2021 January. In addition, the DPRK test fired the new ballistic missile systems such as KN-23 and KN-24 capable of carrying out the nuclear war heads. Other missile systems such as the traditional Nodong and Hwasong series ones are believed to perform the similar tasks. The DPRK is believed to finish constructing the new class of submarines at Sinpo. For the submarines, the new series of Polaris missiles such as 4 and 5 have been shown during the parades.

The short-range missiles and rockets can carry heavy payload war heads, conventional and nuclear. The newly developed KN-23 test fired in 2021 March from Yonpo can deliver the 2.5 ton war head for the given flight distance of around 600 km. Nuclear head capable missile systems with a special functionality such as the shout up and down is a big threat against missile defense systems due to the complexity to properly predict the exact trajectories. It will create the additional development of multi-layer missile defense currently composed of the patriots and THADDs.

The recent progresses at Sinpo is also quite interesting for the potential try-out of a new submarine and the associated SLBM's. The latest demonstration of Polaris-5ㅅ during the January parade attracted the global society in many senses. Firstly, the length of the missile seems to be longer than the previous 3 and 4 series. Secondly, the diameter of a new missile is greater than the previous ones implying that it might carry multiple nuclear warheads. The new Polaris and ICBM's such as KN-27 are not test fired yet. The assure the functionalities of the new products, the new campaigns of launches can be done at any moment. In summary, the DPRK is thought to develop the dual launch systems for both conventional as well as nuclear war heads for short distance and inter-continental targeting. More detailed studies are needed to understand the real capacity of the new systems and the supply chain of the weapon grade nuclear materials to manufacture the war heads.

Keywords: Nuclear weapons, Missile Systems, DPNK, Nuclear Security, SLBM

Modeling of Gas Centrifuge Cascades for Estimating the Production of Nuclear Material

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The denuclearization of the DPRK is indispensable for building a peace system on the Korean Peninsula. Therefore, continuous efforts are needed to achieve the denuclearization of DPRK. As part of these efforts, the Korea Institute of Nuclear non-proliferation And Control (KINAC) is conducting R&D to build a foundation for denuclearization verification. Through this R&D, we will diagnose the technological capabilities for denuclearization and try to build a preemptive non-proliferation regime.

The inspection target needs an appropriate plan at the implemental verification stage, such as preparation for equipment and technology and efficient resource allocation, including human resources. However, when a country or facility where information is difficult to access is subject to inspection, it is necessary to estimate the information in advance based on the theoretical background to facilitate access to the information.

In this respect, the Physical Model can be one of the measures for estimating the information. The IAEA proposed the concept of the Physical Model. However, the Physical Model to be utilized in this study goes beyond this concept and uses a visualized process modeling. We can estimate nuclear material production based on the nuclear fuel cycle facilities in DPRK using an extended Physical Model. Here, the explanation is limited to the uranium enrichment facility in the entire model.

According to Professor Hecker, a DPRK nuclear expert in the United States, DPRK has confirmed that it has already secured an enrichment facility in Yongbyon. According to this, the uranium enrichment facility in Yongbyon has a total of 2,000 centrifuges. However, various cases can happen, including additional facilities, expansion of facility capacity, and centrifuges improvement. Therefore, there is a need for a quantification tool that can reflect these unpredictable changes. In this study, based on the enrichment facility in Yongbyon, we have considered the basic principle of the uranium enrichment process and developed a cascade process model under a specific separation coefficient.

We will estimate the production of highly enriched uranium based on the developed model and prepare the basis for evaluating DPRK's nuclear capabilities. This model will be a part of the Physical Model for estimating nuclear material production, including the DPRK's nuclear fuel cycle.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1905008).

Keywords: Nuclear non-proliferation, Uranium enrichment, Physical model, Discrete Event Modeling

Safeguards Implementation of the Radioactive Waste Containing the Nuclear Material

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In this paper, the IAEA (International Atomic Energy Agency)'s safeguards implementation process was considered when the radioactive waste containing the nuclear material is stored in a nuclear facility or disposed of as final on a disposal facility. Under the ROK-IAEA comprehensive safeguards agreement, all nuclear material is subject to IAEA safeguards implementation so it can be ensured that nuclear material is used only for peaceful purposes. And the radioactive waste containing nuclear materials is no exception for safeguards implementation. KINAC has continued to discuss with the IAEA over the past few years about the safeguards measures on a radioactive waste. First of all, the radioactive waste containing nuclear material generated from the processing or operating which is deemed to be unrecoverable for the time can be reported in terms of 'TW (transfer to retained waste)'. The nuclear material transferred to retained waste can be stored at the facility and is not included in the inventory verification, but continues to be subject to IAEA safeguards. Other studies have shown that concentration for termination of safeguards is based on the concentration of uranium ore. In this regard, if the concentration of nuclear material in radioactive wastes is below the concentration of uranium ore, it may be requested to terminate safeguards implementation to the IAEA or be reported in terms of 'LD (measured discards)' in referring to the relevant 'Facility Attachment'. And, if the concentration of nuclear material in radioactive waste exceeds the concentration of uranium ore, it can be transported to Wolseong Low and Intermediate Level Radioactive Waste Disposal Center (WLDC) for disposal. And this radioactive waste containing nuclear materials should be subject the safeguards implementation by the IAEA, So KORAD, the operator of the WLDC, shall continuously manage this material. Finally, the concentration for safeguards termination and self-disposal were also compared for each radionuclide such as U-235 and U-238 and it was confirmed that there was no significant difference.

Keywords: Nuclear material, Radioactive waste, Safeguards implementation, Termination of safeguards

Radio Active Waste Storage Facility Located in Nuclear Power Plant and Physical Protection

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The spent fuel dry storage should continue to operate after decommissioning the power plant, and what measures should be taken to ensure that there is no gap in physical protection.

KHNP has obtained permission to operate a nuclear power plant under Nuclear Safety Act and is a nuclear licensee under the Act on Physical Protection and Radiological Emergency (hereinafter referred to as “APPRE”). In addition, nuclear facilities under the APPRE include facilities for storing radioactive waste. But nuclear facilities mean the facilities that nuclear licensees are authorized by the government. Permission for operation of nuclear reactors and permission for storage of radioactive waste are different.

Spent fuel storage corresponds to related facility, the handling and storage facility for nuclear fuel materials. In the case of decommissioning a power plant, the related facilities must also be decommissioned to complete the decommission. However, spent fuel storage need to continue operating without decommissioning and physical protection measures should also be applied.

After decommissioning the power plant and the operation permit is terminated, existing operators are no longer nuclear licensees under the APPRE. And spent fuel storage is no longer nuclear facilities. Therefore, physical protective measures under the APPRE can continue to be applied only after obtaining separate permits for spent fuel storage.

When constructing additional MACSTOR in Wolseong, it was considered as a related facility and went through the process of permission to change the operation, not new permission for radioactive waste storage facilities. Physical protection regulations also went through the process of change review, not the initial review.

However, in the case of decommissioning, permission for modification shall not be obtained as permission for operation is terminated. Under the current Nuclear Safety Act, KHNP will have to decide to dispose of spent fuel and obtain permission to build and operate a radioactive waste storage facility. In addition, new physical protection regulations should be prepared to undergo initial examination.

In terms of the Nuclear Safety Act, a new form of permission can be created to operate facilities that store spent fuel intermediate while terminating existing operating permits. In such cases, the concept of nuclear power operators under the APPRE shall also be amended simultaneously to prevent gaps in physical protection.

Keywords: Decommissioning, Physical protection, Licensees, Radio active

Development of Safeguards Technical Report for the Small Modular Reactor in Korea

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An intensive collaboration among IAEA, NSSC, KINAC and KAERI is well under way in order to apply IAEA's safeguards by design (SBD) concept into the small modular reactor (SMR) in Korea, or SMART, and to establish a foundation for SBD to be faithfully implemented from the initial design stage. The safeguards technical report (STR) is a document that IAEA inspectors refer to when verifying safeguards, and its draft version of STR was prepared by adding all the information related to safeguards activities for SMART facilities. This report provided information regarding the implementation of international safeguards that may be taken into consideration during SMART design processes. Introductory information regarding SMART design was briefly provided and detailed routes of nuclear material transfer were investigated in the view point of nonproliferation. Safeguards approach towards SMART design including material balance area (MBA), key measurement point (KMP), possible diversion scenarios, etc. was dealt with in detail.

The first safeguards technical report on SMRs to be published by the IAEA is expected to serve as a model example for the preparation of STRs by other countries. It is also expected to raise awareness of the need for SBD in member states and the nuclear power industry and contribute to cooperation among IAEA member states, as well as contributing to enhancing international credibility and transparency in SMART facilities as the best and the first practice for applying SBD into SMRs.

Keywords: Safeguards by design, Safeguards technical report, Small modular reactor

Development of Real-time Isotope Detection Method for Nuclear Forensics Using Laser-induced Breakdown Spectroscopy

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Nuclear forensics, tracking the origin and history of nuclear materials, is an essential activity to prevent the trafficking and/or illicit use of nuclear materials and to achieve nuclear security and nonproliferation. A rapid and precise detection of isotopes is indispensable for the characterization of forensic target materials. However, the typical methods for isotope detection such as ICP-MS, TIMS, and, SIMS cannot be used for the on-site identification due to their complex measurement procedure, which makes it hard to immediately respond to emergency situations. One of the promising methods to overcome this limitation is laser-induced breakdown spectroscopy (LIBS) by employing the laser-induced plasma as the excitation source. LIBS could not only provide a real-time detection capability regardless of the chemical/physical forms of target materials but also prevent the risk of radiation exposure through stand-off detection.

In the present study, the optical configuration of LIBS was designed for real-time isotope detection. Some isotopes (H, B, Fe, and Sr) with considerable significance in the nuclear industry were examined based on their atomic and molecular emissions from laser-induced plasma. The test samples with varying isotopic abundances in different physical and chemical forms were studied. The spectral difference between isotopes, i.e. isotopic shift, could be clearly observed, whereby isotopes could be distinguished in complex spectra. In order to improve the detection sensitivity, various signal enhancing configurations have been used, such as double-pulse and resonance excitation. Using these methods, the spectral intensity was improved to reach the detection limit of ppm-level by ~10 times. In addition, quantitative analysis of isotopes was conducted by partial least squares regression (PLSR), which is one of the powerful chemometric methods. With the cross-validation, the isotopic abundances of target samples with errors of less than one percent could be precisely assessed. This study demonstrated the feasibility of LIBS for real-time isotope detection and provided scientific insights into the emission behavior of laser-induced plasma.

Keywords: Nuclear forensics, Laser spectroscopy, Isotope analysis, Real-time detection

The Status of Compliance Program Implementation Practices at KAERI

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The Korean government has been controlling the transfer of intangible technology (ITT) under the Foreign Trade Act since 2014. In particular, the increase in international exchanges and foreign employment has led to the need for systemized management of intangible technology transfer, which means the transfer of technology which does not have any physical forms. KAERI acquired Compliance Program (CP) license from the Ministry of Trade, Industry and Energy in 2018. This paper introduces the status of CP implementation practices and proposes effective and specific measures to strengthen the management of intangible technologies transfer at KAERI.

The transfer of intangible technologies includes both domestic and foreign transfers and takes various ways, generally including meeting, lectures, education, presentations and e-mails. After checking the actual work which could cause intangible technology transfer at KAERI, it is confirmed that foreigners visit, overseas working and business trips, cooperation with foreigner experts, and information exchange using e-mail or video conference call with foreigners. In order to prevent illegal transfer of strategic technology to these tasks, regular education or notifications have been performed. However, in this way, it is difficult to fully trust exist practices because the related works cannot be checked in real time and it depends on the applicant's implementation.

The existing administrative work supporting system has been linked to the KAERI Export and Import Control System, and the applicant have to prepare an ‘Export control check-point’. At the same time, in the security evaluation process of the security department in charge of hiring foreigners, export control items are added to the personnel committee to determine eligibility for employment based on evaluations.

As a result, instead of general notifications, separated alerts in the process of each identified task can prevent omission or misunderstanding of work and increases overall efficiency. Therefore, it is possible to respond appropriately to demands for strengthening export control measures related to technology transfer, and to facilitate tracking of past history and establishing related strategies by enabling operators to database CP related information for efficient export control.

Keywords: Compliance Program (CP), Export Control, Intangible Technology Transfer (ITT), Strategic items

Study on Enhancing the Use of Unstructured Open-Source Information for Nonproliferation Monitoring Based on a Text Mining Approach

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Statistical data on the status of Nuclear Armed State's nuclear arsenals and capabilities remain undisclosed since 2009, when Nuclear Armed State refused international verification of its nuclear activities. Hence, collecting and analyzing open-source information is the only available method to monitor its nuclear programme. However, much of these open-source information is comprised of unstructured text data, which typically requires further processing to access the information contained within. This study uses 400 articles collected in 2020 as a data source. Articles include speeches, statement and official newspapers published by the government, evaluation reports of nuclear activity published by international organizations and government agencies, satellite imageries from specialized institutes, and related media articles. The collected information was analyzed monthly and the researcher categorized these into 9 topics based on what they perceived the main topic to be.

This study seeks to find an improved method of unstructured open-source information, focusing on the collection and analysis of information related to Nuclear Armed State's nuclear programme. By reviewing related literatures and analyzing current practice of KINAC, we suggest methods and procedures to determine (1) the appropriateness of KINAC's current classification criteria, (2) trend analysis of topics related to nuclear activities/nonproliferation (hot & cold topics), (3) sentimental analysis for predicting events related to nuclear activities/nonproliferation. Finally, information management format was also analyzed to facilitate proposed text mining approach.

Keywords: Open-source information, Nonproliferation monitoring, Text mining, Topic modeling, Sentiment analysis

1분과

핵주기정책· 규제 및 비확산(Poster)

Case Study of Safeguards on a Small Amount of Nuclear Material

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Under the Safeguards Agreement between the ROK and the IAEA (INFCIRC/236) and the Additional Protocol (INFCIRC/236/Add.1), ROK is subject to IAEA safeguards for all nuclear materials. In particular, not only the major nuclear facilities such as nuclear reactors, and manufacturing facilities using nuclear fuel, but also companies using nuclear material in quantities less than one effective kilogram should undergo IAEA's safeguards inspection.

With respect to scope of nuclear materials subject to safeguards, the Article 34 (c) of IAEA's INFCRIC/236 specifies any nuclear material of a composition and purity suitable for fuel fabrication or for being isotopically enriched shall become subject to the safeguards.

However, these criteria do not present a lower limit on the composition and purity of nuclear material, which makes it difficult to decide whether the material is scope of IAEA safeguards. In particular, small amounts of nuclear material such as thorium welding rods are readily available on the Internet and are very difficult to accounting and control, and the attractiveness of nuclear weapons in terms of safeguards is significantly low.

In this paper, the safeguards and accountancy on small amounts of nuclear materials in Canada is studied. Based on this, direction of improving safeguards for small amounts of nuclear materials in Korea is suggested.

Keywords: Safeguards, Accountancy and control, Composition and purity, IAEA

Development of Optical Fiber Visual Inspector for Spent Fuel Verification

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For CANDU (CANada Deuterium Uranium) reactor type spent fuel storage verification, KINAC (Korea Institute of Nonproliferation and Control) has developed an optical fiber probe system (OFPS) based on radiation measurement. Although most of spent fuels can be verified using the OFPS, several indistinguishable spent fuels can be observed due to the non-uniform burn-up and different cooling periods. Visual inspection using the digital camera is one of the direct verification methods of indistinguishable spent fuels. General CCD type digital camera is not appropriate for inspection due to activation of the sensor and noise by radiation under high radiation environment in the spent fuels storage. Also, the gap between the spent fuel bundles is too narrow (few centimeters) for the camera module to insert the stacks. The visual inspection using an optical fiber is a promising candidate to overcome high radiation and the structure of the spent fuel stacks. In this study, a prototype of optical fiber type visual inspector has been developed as an instrument of assistance of radiation measurement for verification of spent fuels. Multi-core bundle type optical fibers with a light emission lamp to obtain distinguishable images. The developed visual inspector can be used as standalone or with radiation measurement system. The visual inspector has been confirmed the feasibility for direct verification.

Keywords: Spent fuel, Optical fiber, Visual inspector, CANDU

A Study on Application of AI Recognition Technology to Nuclear Import and Export Control

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The recent advances in computer technology have dramatically accelerated the progress of machine learning technology for data and knowledge acquisition and thus AI technology. Substantial efforts have been made to use AI technology to prevent accidents or assist operators, especially in nuclear power plant operation support. However, only a few cases have been reported to date regarding the successful application of AI technology in the nuclear nonproliferation field. The present study aims to identify the types of AI technology available, examine the recent trends of relevant technology development, and further determine which technologies, among them, are suitable to be applied to import and export control.

AI technology can be largely classified into recognition technology, which serves to recognize a given situation; learning and inference technology, which allows the recognized situation to be learned and then predicts what is to come next; and finally acting technology, which takes actions to respond to the predicted situation. This process is quite similar to how humans think and behave, and thus technology development is also following a similar path.

Since it started to develop in earnest in the early 2010s, recognition technology has already been advancing to the point where its ability is no longer inferior to that of humans. This, in particular, is the case for image recognition. In 2015, the image recognition rate of AI technology was recorded at 96.43%, higher than that of humans at 94.9%. In 2017, the figure was even higher at 97.85%.

In import and export control, any businesses or persons who deal with strategic items are potentially subject to regulation. The problem is that it is practically impossible for regulatory authorities to manage and supervise each and every one of these items with their limited human and physical resources. Using advanced AI technology, however, it is possible to extend the scope of regulation and implement regulatory activities in a more efficient and effective manner. Notably, recognition technology is already ready for practical use, being capable of recognizing objects from images faster and more accurately than humans. Most export items, including strategic items, are shipped through airports and ports where they are recognized and verified using X-ray equipment. It will then be possible to determine whether or not given items are strategic items by examining the X-ray images obtained in the process using recognition technology.

Keywords: Export Control, Artificial Intelligence(AI), Recognition Technology, X-ray Image

Study on Application of Regions With Convolutional Neural Networks (R-CNN) and X-Ray Image-Based Object Detection Model to Identification of Trigger List Items

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Most materials, including the Trigger list items, are transferred through airports and harbors, which identify the materials without opening cargo or luggage through X-ray inspection and take necessary actions. Recently, studies have been conducted worldwide on objection identification of not only firearms and drugs but also strategic materials. The present study was conducted to review the various methods for identifying product items and determine which method is appropriate for the trigger list items.

Object detection technology, which detects specific objects or things from given image-based information to identify the positions and types of things, it is classified into 1-stage detectors for carrying out localization and classification simultaneously and 2-stage detectors for carrying out the two tasks sequentially.

The R-CNN model is a two-stage model that carries out localization and classification sequentially. R-CNN, an early technology of object detection, has two major disadvantages. First, the execution time is very long because CNN is performed for each region of the region proposals generated by the localization. Second, the learning in the classification stage, including CNN, Support Vector Machine (SVM) and Bounding Box Regression, is not performed at once because the operations for the individual region proposals are not shared. However, the new R-CNN model, these problems has been resolved, requires just 198ms to perform the CNN process, which takes 50 seconds in the conventional R-CNN model.

The X-ray image-based object detection has two differences compared with the general object detection technologies. First, in X-ray image-based object detection, the data for constituting a database are insufficient. Second, objects overlap each other. At the 2019 IEEE Conference, a new object detection model, developed by applying the class-balanced hierarchical refinement technology, was presented to divide a single image into multiple layers to recognize an object with the three-dimensional shape rather than with the cross-sections.

The development of an actual algorithm requires the consideration of not only the insufficient X-ray images of the trigger list items but also the overlapping of the items at the inspection site. To acquire the necessary data, a simulation was performed by using the Monte Carlo N-Particle (MCNP) code. In addition, due to the constraint that actual cargo X-ray spectroscopy may not be used, small models of the trigger list items were prepared to obtain X-ray images of the objects from various angles and apply the obtained data to the identification algorithm. In addition, the abovementioned class-balanced hierarchical refinement technology will be applied to the obtained images to increase the amount of learning data, thereby improving the performance of the algorithm.

The performance of the object detection algorithm may be improved by constituting the object detection algorithm based on the R-CNN and performing continuous learning by preparing models for acquiring image data and applying the technology to divide an object image into multiple layers. The performance improvement will lead to the development of an identification algorithm that is applicable to on-site inspection.

Keywords: R-CNN, Object Detection, X-ray image, Trigger List Items

A Development of Integrated Export Control System for Improvement of Nuclear Strategic Items Control

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The international community has standardized export control of strategic materials to prevent the spread of weapons of mass destruction (nuclear, biochemical, etc.), and Korea has strictly followed the procedures of classifying strategic items under the Foreign Trade Act. With the development of technology and growing international cooperation, Korea has established more elaborated and specific export control policies, which can control the transfer of strategic items generated in various ways. For applicants to export items or technologies, it is important to follow-up these policies quickly and accurately.

Export control of strategic items is led by government organizations such as the Nuclear Safety and Security Commission and the Ministry of Trade, Industry and Energy. In cases of items on trigger list, it is judged by Korea Institute of Nuclear Nonproliferation And Control and the applicant uses the web system, the Neps. On the other hand, in cases of nuclear dual-use items, Korea Security Agency of Trade and Industry determines them and the applicant shall use the web system, the Yestrade. As previously stated, because it is referred to each agency, the applicant's documentation is unnecessarily duplicated on some entries, such as information on basic items, purpose of end-use or trade information. At the same time, it is difficult for applicants to easily grasp various tasks related to export control.

Therefore, it is necessary to develop a specialized system that supports timely transfer of strategic items and helps applicants understand as well as enables tracking and management of data. This paper proposes measures to improve export control of operators by introducing the KAERI Export and import Control System (KECS) development process and key results. The scope includes all import and export control tasks of strategic items at KAERI, and the applicant and the person in charge are able to control and follow-up the progress in real time. Also, the system encompasses the management of intangible technology transfer and compliance program operations and control. As a result, it is intended to establish an import-export control system that integrates related implementation tasks to increase the efficiency of the applicant's work and reduce administrative burden.

Keywords: Strategic items, Export Control, Management System, Nonproliferation

The Nuclear Industry Information Collection and Analysis System

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As overseas exports of nuclear power plants have diversified, there is a possibility of illegal transfer of strategic materials from companies that handle trigger list items because they do not recognize export control regulation. Therefore, the possibility of illegal exports should be minimized by preemptively discovering the nuclear-related companies and incorporating them into the export control system. Research on the nuclear industry information collection and analysis system was conducted to periodically search for nuclear-related industries that are not currently incorporated into the export control system. The information collection system conducts data crawling of nuclear-related industries and the collected information is analyzed to determine whether the company handles strategic materials in the analysis system.

In order to collect information on the domestic nuclear power plant industry, the initial database was established by collecting information related to suppliers of major nuclear power plant construction. The list of similar companies was obtained from NICE's corporate information homepage by utilizing the company information in the initial database. The secured similar companies are searched on the web portal to find the company's website and extract all the corporate information and text through web crawling. This extracted information is stored in the database and sent to the analysis system for determination of whether it is related to trigger list items.

Rule-based classification was used to determine whether the collected companies handle strategic materials. Five classification conditions have been employed, and if any, they are classified as strategic material handlers. This is to prevent companies from being omitted even if the classification is less accurate. The five classification conditions consist of comparing the consistency of strategic materials, item classification, control technology, and classification technology and cosine similarity with the description of trigger list items.

Using web crawling and rule-based classification, the nuclear industry information collection and analysis system was established. The developed system will be used to discover export control outreach targets in connection with NEPS (Nuclear Export and Import Control System) in the future.

Acknowledgements

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Keywords: Export Control, Crawling, Classification, Strategic Item

Study on the Improvement of Strategic Technology Transfer Management for Nuclear Plant Technology Export License

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Korea has joined all four international export control systems (NSG, AG, MTCR and WA) to prevent the spread of Weapons of Mass Destruction (WMD), and controls exports by reflecting the controlled items and guidelines set by each system. To export Trigger List items based on NSG guidelines, the exporter must obtain an Export License of Strategic Item from the government. Export control is needed because WMD development is virtually impossible with one country's power, mainly through trade and cooperation between countries.

South Korea won four nuclear power plants in the UAE in 2009, and has been under construction for 10 years since then. Among the various technical documents required for construction, documents related to the Trigger List have been transferred individually after obtaining the Export license.

Individual Export licenses usually take more than 15 days and up to 30 days, making it difficult to send data urgently according to the construction schedule. Accordingly, the nuclear plant technology export license was introduced in 2014, which collectively permits nuclear strategic technologies included in the large-scale nuclear plant export during the project period. However, in the case of technology transfer, it is still necessary to determine whether or not the technology is strategic from a specialized agency.

Therefore, for more efficient strategic technology management, if the exporting company has the system and ability to manage strategic technology, the method for the exporter to determine whether or not the transferred technology is a strategic technology on its own is proposed. However, it is necessary to supplement operational problems through regular inspections during the project period to confirm the continuation of the management system and capabilities, and the results of strategic technology decisions.

When permitting nuclear plant technology export, it evaluates whether the company's internal regulations for strategic technology management, organization, and management system are established, grants the authority to determine strategic technology, and reports the results of strategic technology determination to the government every quarter. Similar to the physical protection inspection of nuclear facilities, the government can regularly inspect during the license period to request corrective action and decide on the maintenance or suspension of licenses according to follow-up measures.

As such system improvement enables the urgent transfer of strategic technologies within the license when exporting large-scale nuclear power projects, it is expected that exporters will be able to smoothly continue their export business while implementing the export control system.

Keywords: Nuclear Plant, Strategic Technology, Export License, Inspection, Urgent transfer

Study on Need to Introduce Korean Obligation Code for Managing Nuclear Materials Subject to Nuclear Cooperation Agreement

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The production of nuclear materials used in Korea includes various physical and chemical processes. The countries participating in each of the processes export the processed nuclear materials for an economic purpose. To prevent the diversion of the exported nuclear materials to nuclear weapons, various obligations are applied to the importing countries, including advance consent of retransfer, through nuclear cooperation agreements.

Imported nuclear materials are subject to obligations based on agreements with individual countries through which the materials have passed for the processes; the obligations of all countries through which the materials have passed are applied. For example, when a nuclear material is mined in Australia, converted in the US, and enriched in the UK, the agreement obligations of Australia, the US and the UK are all applied to the material. These obligations imposed on nuclear materials are called foreign obligations.

Some countries, such as the US and EU, have established and applied obligation codes to indicate and manage foreign obligations of a single or a plurality of countries that are applied to nuclear materials. The obligation code, a combination of a single-digit or double-digit number and a letter, has many advantages in managing nuclear materials subject to the Agreement in accounting, but it has not yet been introduced to Korea.

When obligations codes suitable for Korea are established and applied to accounting, the agreement obligations of individual countries that should be fulfilled in the retransfer of nuclear materials may be easily tracked and verified, and the inventory for each country, which should be reported to Canada, Australia and the US each year, can be easily investigated.

For nuclear materials to which foreign obligations are applied, the applied obligations may be mutually swapped according to the principles of substitutability, proportionality and equivalence. For example, in retransfer, nuclear materials with many obligations may be left in the inventory in Korea or those with many obligations may be exported preferentially depending on the state of export. In this case, the application of obligation codes is necessary to track and manage the obligations applied to the nuclear materials.

The agreement obligations applied to nuclear materials become more difficult to track as time passes. This paper proposes to apply obligations codes suitable for the domestic environment as early as possible to effectively manage the import and export of the nuclear materials subject to the agreement through retransfer.

Keywords: Nuclear Cooperation Agreement, Nuclear Material, Retransfer, Foreign Obligation, Obligation Code, Export and Import

Importance of Physical Protection - Cybersecurity Interface and Development of Related Training Program

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The word physical protection – cybersecurity interface may be an awkward word in the field of nuclear security. We often use the term security – safety interface. In fact, physical protection – cyber security belongs to nuclear security, and because the goal of each system are same, without needing to mention interface. However, as the requirements for each system increased and the licensee became independent organizations, the time to rethink the interface came. In the general industrial security field, physical security and cybersecurity have been developed by being divided into two types, but synergies that appear when the two are interfaced the name of integrated security are being talked about.

Nuclear facilities take various measure to prevent sabotage or unauthorized removal, and a set of these measures is called a physical protection system. Therefore, in can be seen that cybersecurity measure with the same purpose are also included in the physical protection system. However, this misunderstanding made it necessary to interface two because we think that preventing only physical attacks is physical protection. There are various threats that cause sabotage or unauthorized removal, and these threats include cyberattacks and complex attacks including cyberattacks. Anyone who wants to sabotage or unauthorized removal will not differentiate between physical attacks and cyberattacks, but will only consider various strategies to achieve goal. The same goes for those of us who need to protect it. Rather than distinguishing the two, all actions should be carried out in terms of achieving the goal. The intruder is like a highly intelligent disaster. They will never try to cross a high wall, and they will never try to break a thick cyber firewall. Always keep in mind that the attack is made up of vulnerable parts, you should perform a balanced protection.

We intend to develop training programs to improve this perception. The first step is a practice-based training program in which both physical protection and cybersecurity officers can participate and cooperate to solve problems. We will share each other's perspectives for solving problems and show that correct actions can be carried out only when actions in the two areas are carried out in an integrated measure. Therefore, as the subject of training, we intend to carry out "Cybersecurity training based on physical protection related equipment". We will show how cyberattacks and affects physical protection.

This paper summarizes the importance of physical protection – cybersecurity interface in the nuclear field and proposes a training program that can have this awareness.

Keywords: Physical protection, Cybersecurity, Interface, Training, Education

Preparation for Domestic Regulation Standard Related to Safeguards Implementation of Nuclear Facility in Decommissioning

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The Korea Hydro & Nuclear Power (KHNP), determined the decommissioning of the Kori unit 1 in 2015. According to this determination, the operational status of the Kori unit 1 was changed from normal operation to permanent shutdown, and the authority of the ROK officially declared the decommissioning of the Kori unit 1 in 2017. The regulatory body of the ROK has been preparing the regulation guidelines in response to the related demands. These regulatory demands include not only nuclear safety but also nonproliferation (Safeguards) aspect. Specifically, the International Atomic Energy Agency (IAEA) has been developing the Safeguards guideline for a nuclear facility in decommissioning. This guideline includes detailed contents for nuclear material removal, essential equipment dismantlement, provision of information to the IAEA, and IAEA inspection. Among them, the provision of information to the IAEA includes nuclear material accounting and control reports and design information questionnaire (DIQ). The present DIQs of the domestic nuclear power reactors do not contain detailed information related to decommissioning plans. Therefore, the IAEA revised the general DIQ form to include the post-operation information consisting of decommissioning schedule date, general decommissioning plan, removal and recovery of nuclear material plan, and removing and rendering inoperable of essential equipment plan. Therefore, the DIQ of the Kori unit 1 will be updated to include them. Any change in the design information of a nuclear facility should be reported to the IAEA by the Article 45 of the Comprehensive Safeguards Agreement between the IAEA and ROK. This requirement is also included in the ROK legal frame under the NUCLEAR SAFETY ACT. Therefore, any change related to the decommissioning plan of a nuclear facility will be reported to the IAEA by the DIQ revision. However, more detailed guidelines for a nuclear facility in decommissioning is necessary in the domestic regulation frame. For example, retained waste should be re-transferred to book inventory, and a nuclear facility should submit null nuclear material accounting and control reports although there is no nuclear material. Additionally, there are several important issues that should be considered relating to the essential equipment dismantlement, the IAEA inspection, etc. Therefore, the technical supporting organization of the Nuclear Safety and Security Commission has been developing a proper domestic regulation standard for a nuclear facility in decommissioning. Conclusively, this regulation standard under development will provide appropriate guidelines to a nuclear facility and support to keep the obligations resulted from the Comprehensive Safeguards Agreement.

Keywords: Safeguards, Nuclear Facility, Decommissioning, Regulation Standard

Analytical Method to Determine the Mass of Uranium Based on Active Neutron Multiplicity Counting

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Various detection techniques based on single, double, triple neutron multiplicity counting are widely used for verification of quantitative analysis of special nuclear materials since their total mass in a sample is proportional to neutron population in a time frame. Since spontaneous fission from special nuclear materials is rarely occurred, the active neutron interrogation method employing an external source is practically used to increase the neutron multiplicity distribution.

A numerical method based on the active and passive correlated technique was proposed in order to determine the mass of uranium in a sample by employing a D-D neutron generator and He-4 scintillation detector array. The mass of uranium equation was basically derived from the neutron multiplicity point-model equation corresponding to the single rate of the spontaneous fission neutrons, (α, n) neutrons, induced fission neutrons, and interrogation neutrons come from the D-D generator.

The energy threshold was adopted to distinguish only fission neutrons since fission neutron spectrum is dominant at energies greater than 3 MeV. Therefore, the equation was simplified by cutting off the term of (α, n) neutrons and interrogation neutrons, and then re-written in terms of the interrogation yield (Y), the ratio of passive to active neutron population (R), fission rate (F), and fission multiplicity (ν).

Considering the abundance and fission cross-section of uranium isotopes at 2.45 MeV, the terms F and ν in the equation become to only depend on U-235 isotope since the fission cross-sections of other uranium isotopes are relatively smaller than U-235. Finally, the mass of U-235 can be calculated by considering the time change of neutron population between D-D generator pulses.

Based on the derived uranium mass equation, it was concluded that the mass of U-235 can be directly calculated by using 2.45 MeV interrogation neutrons. The next step is 1) to verify the proposed method by a proof of concept experiment, 2) evaluate relative uncertainties, and 3) carry on the parametric studies to optimize the experimental setup.

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Keywords: Safeguards, Active neutron assay, Special nuclear material quantity, Energy cut-off, Numerical methodology

An Analysis of Training Requirements by DACUM Method in the Course for Fuel Cycle-related Researchers of NNT

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The Korean government announced the "Four Principles on the Peaceful Use of Nuclear Power" in 2005 and established “Korea Institute of Nuclear nonproliferation and Control” (KINAC) as a specialized organization for nuclear nonproliferation activities in Korea including Nuclear Nonproliferation Training (NNT).

The NNT is compulsory programs by laws for nuclear fuel cycle-related project managers and the personnel for nuclear material accountancy in nuclear facilities according to Nuclear Safety Act. It consists of two courses, the course for personnel in charge of material accountancy and control of Special Nuclear Material (SNM) at nuclear facilities and the course for project managers of nuclear fuel cycle-related research. The details of the courses are 1) International Treaty and Regime on Nuclear Nonproliferation, 2) Safeguards & Export Control, 3) Current issues on Safeguards & Export Control, etc.

KINAC, as an organization entrusted by Nuclear Security and Safety Commission (NSSC) for NNT, studied the contents for nuclear fuel cycle-related researchers to improve the education effect. The analysis tool was the DACUM (Deveoping A Curriculum) method which is an effective process that incorporates the use of a focus group in a facilitated storyboarding process to capture the major duties.

Three responsibilities (A. research related to nuclear fuel cycle, B. report on export and import control of international regulated goods, C. support for IAEA inspection) and four related tasks (A-1. Annual Report for Additional Protocol, B-1. Nuclear Material Export and Import Approval, B-2. Application Process for Export and Import Control of Internationally Controlled Materials) were concluded by the interviews with the trainees. Based on these results, eight subjects needed for trainees were derived by grouping necessary knowledge belonging to four tasks. The eight subjects are as follows; 1) The Need of NTT, 2) Case-Based Report on Nuclear Fuel Cycle-related Research, 3) International Nuclear Regime and NTT, 4) Report on Nuclear Fuel Cycle-related Research, 5) Support on IAEA Inspection 6) Report on Internationally Controlled Materials 7) Export Control on Internationally controlled Materials (Introductory Course), 8) Export Control on Internationally controlled Materials (Advanced Course. These subjects will be applied to the courses step by step until 2023.

This research is the first analysis conducted for the NTT course for project managers of nuclear fuel cycle-related research since the implementation of NTT in 2005. Through the implementation of legal education based on the needs of trainees, KINAC will continue to raise the knowledge and awareness of nuclear nonproliferation among the trainees.

Keywords: NNT (Nuclear Nonproliferation Training), Project managers of nuclear fuel cycle-related research, Compulsory program by laws, Education, DACUM (Deveoping A Curriculum)

Comparision of Penal Provisions on Sabotage for Radio Active Waste Facilities

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Article 39 of the Radioactive Waste Management Act stipulates that any person who destroys or operates radioactive waste management facilities with malicious intention shall be punished. Meanwhile, the Act on Physical Protection and Radiological Emergency (hereinafter referred to as "APPRE") also has a regulation on the punishment of those who commit sabotage (Article 47). This report will compare these two provisions.

According to the Article 2①.5 of the APPRE, sabotage is an act of destroying nuclear facilities or obstructing normal operation of them. Nuclear facilities herein refers to facilities for storing, processing and disposing of radioactive wastes.

The management of radioactive wastes also means the shipment, storage, process and disposal of radioactive wastes (Article 2 of the Radioactive Waste Management Act). Therefore, destroying or wrongfully operating radioactive waste management facilities is included in the concept of sabotage in the APPRE.

The Radioactive Waste Management Act (with the article 39) was enacted in March 2008. But, the APPRE also already had a regulation to punish sabotage when it was enacted in 2004. Although the APPRE was revised in 2014 to reflect the contents of the Convention on the Suppression of Nuclear Terrorism, the clause on punishment of sabotage has been in place before, and the contents have not been affected by the amendment.

The concept of sabotage in the APPRE requires the release of radioactive materials or exposure to radiation. However, the Radioactive Waste Management Act does not mention radioactive material emissions or radiation exposure. But this act should also be interpreted as punishing only those who emit radioactive materials.

The crime prescribed by the Radioactive Waste Management Act also may be punished by the APPRE. The Radioactive Waste Management Act is under the jurisdiction of the Ministry of Trade, Industry and Energy and the APPRE is under the jurisdiction of the Nuclear Safety and Security Commission. The two different ministries being in charge of each act seems to have resulted in overlapping punishment regulations.

Keywords: Radioactive Waste Management Act, Act on Physical Protection and Radiological Emergency, Sabotage, Punishment

Analysis of Anti-Sabotage Measures for Nuclear Facilities

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If sabotage occurs on nuclear facilities, the impact on society will be enormous, irrespective of the radiation effects. Accordingly, in nuclear facilities, various sabotage prevention measures are configured to prevent sabotage in advance, and efforts are being made to maximize the performance of sabotage prevention measures through appropriate training and maintenance.

Countermeasures against sabotage can be organized into strategies, tools, and human in a large category. First, the strategy can be changed in a number of forms depending on various factors such as the number, goal, location, and time of the impurity who wants to generate sabotage. Accordingly, the strategy may change significantly depending on the capabilities of the commander of the facility. Second, there are various types of tools, such as firearms, vehicles, sensors, fences, etc. that the facility has. Maintenance to ensure that the tool works properly and mastery of the tool by the user of the tool is very important. Third, people can be classified into facility workers who respond indirectly, such as security police, security guards, and reserve troops who respond directly.

This study attempts to identify various anti-sabotage countermeasures that are currently being studied or developed, and suggest countermeasures against sabotage that should be provided in domestic nuclear facilities, focusing on cases of terrorism abroad.

Keywords: Sabotage, Anti-sabotage, Nuclear facilities, Countermeasures, Terrorism

Force-on-Force Exercise and Evaluation for the Prevention of Sabotage in Nuclear Facilities

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Nuclear facilities in Korea are evaluated by NSSC (Nuclear Safety and Security Commission) every year to prevent sabotage, and KINAC is entrusted with training evaluation work. Force-on-Force Exercise is conducted as a two-way free start training in which adversaries with the purpose of sabotage of nuclear facilities suddenly infiltrate. In addition, as part of an effort to create a training environment similar to the real world, KINAC (Korea Institute of Nuclear Nonproliferation and Control) has been conducting training evaluation by introducing MILES (Multiple Integrated Laser Engagement System) and training evaluation equipment from 2016.

In the training evaluation equipment, various training information can be checked. In particular, it has the advantage of being able to check the location information of training participants in real time such as adversary, security police, and security guard. However, research on how to use this for quantitative evaluation was insufficient. As a result of the study on the continuous training evaluation method, it was confirmed that the weak point of protection in the nuclear facility could be identified by measuring the distance between the adversary and the sabotage target targeted by the adversary over time. KINAC (Korea Institute of Nuclear Nonproliferation and Control) developed quantitative evaluation software to actively utilize this for Force-on-Force Exercise evaluation, so that adversary time-distance graphs could be easily derived.

In this study, the time-distance graph of Force-on-Force Exercise was derived using quantitative evaluation software, and the training results of each nuclear facility were compared and analyzed to verify the quantitative training evaluation method using the time-distance graph.

Keywords: Force-on-Force Exercise, Evaluation, Sabotage, Adversary

The Status of Expanded Declaration in Accordance With Additional Protocol at KAERI

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The agreement between the government of the ROK and the IAEA agency for the application of nuclear safeguards was signed in October 1975 and entered into force in November of the same year. The ROK signed on Model Protocol Additional to the safeguards agreements (AP) with IAEA in 1999 and the agreements were entered into force in February 2004. Therefore the ROK submitted the initial expanded declaration which includes further information to be required from IAEA under AP. The declaration has included information such as activities related to nuclear fuel cycle, all building on site and future plans etc.

KAERI has submitted the declaration to KINAC by end of March, every year. Usually, the expanded declaration of KAERI has included Article 2.a.(i), Article 2.a.(iii), Article 2.a.(iv) and Article 2.a.(x). The information of Article is following:

- Article 2.a.(i) : Government funded nuclear fuel cycle-related R&D activities not involving nuclear material
- Article 2.a.(iii) : Description of building on site and site map
- Article 2.a.(iv) : Manufacturing activities specified in Annex I
- Article 2.a.(x) : Ten-year plans for the development of the State's nuclear fuel cycle

The first expanded declaration reported in 2004 included all R&D related to nuclear fuel cycles that did not use nuclear materials. Therefore, number of 2.a.(i) had the highest year. After that, only if the information or content of projects is changed like as completion or launch of project, it is reported.

In the case of 2.a.(iii), all buildings, temporary buildings and containers within the KAERI site are reported as well. Even if a building is under construction or construction is completed, changes are also reported. The Temporary buildings is defined that structure has less mobility and is reported the changes. On the other hand, containers are the most volatile structure within KAERI site. In particular, if new buildings are constructed, containers may be installed to construction offices. Therefore, information of all container is reported because of large changes each year.

In case of 2.a.(iv), an average of about 1 report was reported by 2014, but now it is not reported. Also, Article 2.a(x) means a long-term plan, the number of reports is very small and it is about 1 case on average.

Keywords: IAEA safeguards, Additional Protocol, AP, Expanded Declaration

Measurement of Gadolinium Isotopes Using Laser Ablation and Laser Induced Fluorescence

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The measurement of uranium (U) enrichment is important in non-proliferation aspects, since the abnormal U enrichment can indicate that the U is intended to be used for the nuclear weapon or explosive. Usually, the U enrichment is measured with mass spectrometry. Optical spectroscopy has the possibility to be used for the on-site measurement of U enrichment. Due to the safety regulation, gadolinium (Gd) was used as a surrogate of U at present considering the chemical and atomic spectroscopic aspects. The Gd isotopes was measured with Laser Induced Fluorescence (LIF) combined with laser ablation. The Gd sample was placed in a metal chamber in which the ambient gas type and gas pressure were controlled. A Nd:YAG laser beam was incident vertically on the sample to generate laser ablation, and laser beam from an External Cavity Diode Laser (ECDL) was incident parallel to the sample surface to generate fluorescence. The wavelength range of the ECDL was from 405 nm to 406.5 nm, and Gd lines related to fluorescence in this wavelength range were 405.364 nm, 405.472 nm, and 405.822 nm. The emission spectrum from laser ablation and LIF was measured using Echelle spectrometer and EMCCD. Before the actual measurement using Gd sample, a Gd hollow cathode lamp was placed in the location of the metal chamber. Laser beam from ECDL was incident in the Gd hollow cathode lamp and the spectrum of the emission light from the hollow cathode was measured while changing the wavelength of the ECDL. Gd isotopes could be identified from our measurement, and this shows the possibility that LIF can be used for the measurement of U enrichment as the non-proliferation technology.

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Keywords: U Enrichment, Laser Ablation, LIF, Gadolinium

An Estimation of Fissile Material Production Based on Operating Scenario Assumptions From the 5 MWe Yong-Byon Graphite-Moderated Reactor by MCNP6 and SCALE6 Calculation

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The estimation of the fissile material production from 5 MWe Yong-byon graphite-moderated reactor is important to evaluate North Korea's capability to produce nuclear weapons. The amount of fissile material according to each operating period of the Yong-byon graphite-moderated reactor were estimated by many experts, based on the operation histories and the core design of the reactor given in the open literatures. However, in order to conservatively estimate the production of fissile material, assumptions about the operation scenario of the Yong-Byon graphite-moderated reactor are needed.

In this work, a nuclear fuel reload scenario during the operation period of the Yong-Byon graphited-moderated reactor was assumed to calculate fissile material production by MCNP6 and SCALE6 calculation. It was assumed that the reloading scenario used an in-out loading pattern and an out-in loading pattern. The reload period assumed that the fuel was replaced without shutdown after operation for one year. Accordingly, since the longest operating period for each period of the Yongbyon reactor is 4 years, a total of three fuel reloads were assumed. In addition, in that scenario, variables not provided in the open literatures: boron contamination change, power change, and graphite moderator density change, were used for estimating production of fissile materials. Also, the estimated plutonium production for each variable were compared with the expected mass of plutonium according to the operation records. Furthermore, inventories of the Weapon-Grade (WG) plutonium were estimated and the two results of MCNP6 and SCALE6 calculation were compared.

The results from SCALE6 and MCNP6 show that the difference in WG plutonium production for each variable is within 1% at the longest operating time before removing spent fuel. In addition, it was confirmed that the values calculated by SCALE6 and MCNP6 were in the range of the total amount of plutonium produced from 1986 to 2015 during the 5 MWe reactor operation period estimated by Albright.

Keywords: Reload scenario, Fissile material, In-out loading pattern, Out-in loading pattern

Development of Active Seal on Real-Time for Safeguard Goal

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There are generally two kinds of seals - passive and active seals. Passive seals are typically inexpensive to apply and interrogate. But, they may require forensics to determine the seal`s integrity. Also, they do not provide timely detection and reading and inspection are required proximity. On the other hand, Active seals have several advantages such as timely detection, reduced personnel exposure, increasing the complexity of adversarial attack through sophistication, and typically provides a broader suite of sensing. But, they have disadvantages like a typically higher cost than passive devices, to require a system for interrogation and periodic maintenance.

This research is for designing and producing real-time active containment device. The containment is consisting of a receiver and a transmitter and the transmitter is an actual containment device, and the receiver is a wireless receiving device for sending integrity signals and unauthorized access alarms from the containment device to the server. The receiver was equipped with a 7-channel Lora receiving module, and used a micro-controller capable of decoding including AES256. The transmitter is designed with low power, and has a 3.7V ion battery, 3500 mA x 2, and has a solar charging function to increase life time when used outdoors. It uses a Lora transmission module and detects opening/closing and approaching using optical pulse check, opening/closing detection sensor & reed switch, and MEMS sensor.

Currently, we are preparing the environmental test and self-performance test of the containment device, and we are developing an algorithm according to the approach scenario.

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Keywords: Seal, Translator, Lora, Active, Real-time

Uncertainty in the Economics of the Nuclear Fuel Cycles

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Economics of the nuclear fuel cycle system is one of the criteria to be assessed for the comparison of alternatives among the nuclear fuel cycle systems. The life cycle cost of certain nuclear fuel cycle can be quantified in terms of the levelized cost of electricity (LCOE), which is a total life cycle cost of the system divided by the total electricity generation from the system. In the LCOE analysis, there are lots of uncertainties in the unit costs and other parameters that we assumed are involved in the economic evaluations. Especially, cost for the unbuilt future reactors and fuel cycle facilities contains a large degree of uncertainty. Therefore, uncertainty analysis on the life cycle cost should be conducted in the economic evaluation of the nuclear fuel cycle systems.

In the economic evaluation of the nuclear fuel cycle system, the uncertainty in the LCOE can be estimated from the uncertainty of the input unit costs because uncertainty of the LCOE in terms of standard deviation is a consequence of the uncertainty intrinsic in the input cost data. The probability density function of each unit cost for each fuel cycle stages can be taken from the Advanced Fuel Cycle Cost Basis Report which is a publicly available collection of cost data for most nuclear fuel cycle stages.

In this study, statistical test was used for the comparison of two different fuel cycle systems whether one system differ from the other system significantly or not, based on the uncertainties of two LCOE distributions. With the given statistical mean and standard deviation of the distributions from the test case, two-tailed tests for the hypothesis testing (Null hypothesis (H0): 'cost for System A' = 'cost for System B', and alternative hypothesis (Ha): 'cost for System A' ≠ 'cost for System B') were conducted. The p-value, which is the observed significance level, can be calculated from the observed mean and standard deviation assuming that the hypothetical distribution is a normal distribution. When the significance level (α) is 0.05, which corresponds to a confidence level of 95%, the p-values are greater than the significance level (α) and, therefore, the null hypothesis (H0) is accepted.

Keywords: Uncertainty, Economics, LCOE, Statistic

Investigation of the Effect of the GIS Thematic Map on the Segmentation Quality to Improve the Change Detection Accuracy for Countering Nuclear Proliferation

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As high-resolution satellite images can be acquired with the development of satellite sensors, satellite imagery has been widely applied to various fields such as environment, disaster, national defense, nuclear nonproliferation, etc. From the perspective of nuclear nonproliferation, it is very important to detect even small-scale changes, e.g. the steam generation, the cargo movement, and the building extension, which can be regarded as evidence of suspicious nuclear activities. However, when detecting small-scale changes with high-resolution satellite images, a lot of noise in the acquired high-resolution satellite imagery can cause an error in the change detection result. Therefore, many previous studies have been performed on the object-based change detection (OBCD), which can reduce the noise of pixels by grouping pixels with similar spectral characteristics. The process of OBCD consists of the following two steps: (1) the segmentation of the multi-temporal images, and (2) the change detection by overlaying the pixel-based change detection result with the segmentation result. In particular, since the segmentation errors which can occur in the first step can be inherited to the accuracy of change detection, it is necessary to minimize the errors by considering additional options such as importing GIS thematic map for guide layers and adopting the optimized scale factor calibrated by the estimation of scale parameter (ESP).

Thus, in this study, the effect of the GIS thematic map for buildings adopted in the segmentation step was investigated by comparing the segmentation qualities in accordance with or without the GIS map. In detail, the segmentation quality (goodness) was quantitatively investigated with estimating the area-based accuracy indices such as Under-Segmentation (US), Over-Segmentation (OS), and the Quality Rate (QR). As a result, when considering the GIS map, the segmentation quality showed the significant improvement in terms of those three indices. In addition, the segmentation result with the GIS map qualitatively described that segments for the buildings and the others were more obviously classified in satellite images. In conclusion, in order to improve the change detection accuracy related to reducing the segmentation error, the GIS map was applied in the segmentation process of the pixel-object fusion change detection algorithm developed by KINAC in support of human interpretation for satellite imagery.

Acknowledgements

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Keywords: Object-based change detection, GIS thematic map, Segmentation quality (goodness), Area-based accuracy indices

Preliminary Study on the Change Detection of Uranium Tailing Piles With Utilizing Target Detection Results for Nuclear Nonproliferation

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From the perspective of nuclear nonproliferation, it is difficult to directly investigate suspicious nuclear activities in the restricted access areas such as rogue states by the on-site inspection (OSI). Nevertheless, owing to the development in spatial resolution of satellite sensor in recent times, even small changes related to the nuclear proliferation can be detected by remote sensing with high-resolution satellite imagery. Notably, in the case of the uranium mine, uranium tailing piles were observed in lots of satellite imagery, which was generated as a by-product of the mining and milling process. Therefore, if the change of those uranium tailings is detected, the existence of uranium mining activity can be estimated. Many previous studies are researched on target detection for minerals of interest, e.g., limestone and coal, by utilizing similarity indices of SAM, SID, and NS^3.

Thus, this paper performed the target detection algorithm to extract pixels for the uranium tailings from satellite imagery in different times. The change of the uranium tailings was also detected by comparing the target detection results. First, the target detection results for uranium tailing piles in different times were mapped by calculating the similarity (NS^3) between the spectral information of the actual uranium tailings and that of each pixel in satellite imagery. Second, the change of pixels indicating the uranium tailings was extracted in comparison with those target detection results using the pixel-based change detection, i.e., the multivariate alteration detection (MAD). Further, the binary map for the change and the no-change were made by applying statistic thresholds such as 95 % and 99 %. In conclusion, this study suggests the availability of the change detection method for minerals of interest by comparing the target detection results. As a future work, the target detection maps for uranium tailings will be developed to continuously monitor suspected nuclear activities.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 1905009).

Keywords: Uranium tailing piles, Target detection, Similarity, Change detection, MAD

A Study on the Threat Assessment of Domestic Nuclear Facilities by Establishing Peer Groups Using OECD Member States

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The Korean government shall assess threats to nuclear facilities once every three years in order to formulate measures for the physical protection of the nuclear facilities including radioactive waste storage, processing, and disposal facilities under Article 4 of the Radioactivity Prevention Act and Article 7 of the Enforcement Decree of the same Act. The threat assessment on nuclear facilities should be objective and precise for the nuclear security because Design Basis Threats are identified based on the results of the threat assessment. The Korea Institute of Nuclear Nonproliferation And Control tried to organize the peer group by selecting certain countries under similar circumstances with South Korea to develop more objective threat assessment methodology. The progress focused on the identification of national attributes with significant correlations to terrorism and organize the peer group of countries with similar national attributes. However, specific national attributes with significant correlation was not identified. It lead to the conclusion that the occurrence of terrorism depends on the complex characteristics and circumstances of the state rather than on particular attributes.

This study derived various attributes related to terrorism by analyzing the causes and forms of terrorism, and investigated international organizations that demand such attributes as evaluation criteria to the states which want to join the organization. The Organization for Economic Cooperation and Development (OECD) demand most of the identified attributes above for the membership, and thus the study established the threat assessment peer group using OECD member state. However, the study excludes OECD member countries which have a different current circumstances with South Korea. The study finalized the peer group by adding countries with very similar environment to Korea among non-OECD member countries.

This study represents the methodology to develop more precise and objective threat assessment compared to the way to analyze all terrorism incidents, and it could contribute to increase the efficiency of the national physical protection system and resources.

Keywords: Threat assessment, Design basis threat, Physical protection measures, Peer group

A Study on the Nighttime Evaluation Plan for Physical Protection Exercise

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Terrorism against multi-use facilities can be regarded as the most vulnerable in event time when large numbers of people gather. On the other hand, nuclear facilities have the greatest threat of terrorism aimed at sabotage, so nighttime may be vulnerable. Therefore, it is effective to conduct most of the protective emergency drills at night. When training is conducted at night, there may be various factors and evaluation methods for the night training evaluation.

First, there is a method of configuring the evaluation committee and the evaluation target as one-to-one or one-to-many. In most cases, one-to-one can perform an accurate evaluation, but there are many situations that are difficult in reality or cost. One-to-many are affected by the competency of the evaluator, and have the advantage of selecting a major evaluation site and accurately evaluating the part.

Second, the composition of the evaluator can be made up of protection workers in the same occupation, and in this case, there can be an advantage that can complement each other. Objective and accurate evaluation can be achieved when the evaluation center is designated as an external agency or related expert. However, it may be realistically difficult to form a one-to-one evaluation committee.

Finally, there may be tools for nighttime evaluation where it is difficult to secure visibility. You can use a flashlight, but it can interfere with normal training. Accordingly, a thermal imaging camera or night vision can be used.

In this study, various factors and evaluation methods for nighttime training evaluation are investigated, and methods to supplement and develop them are to be studied.

Keywords: Nighttime Evaluation, Physical Protection, Evaluation

Considerations for Developing a Strategy for Japan's Decision to Release Fukushima Water Into the Ocean

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On March 11, 2011, a 9.0 earthquake and tsunami occurred in the Pacific Ocean in the Tohoku region of Japan, and an accident occurred at reactors 1-4 of the Fukushima No. 1 nuclear power plant. Eventually, reactor units 1-3 evaporated all the cooling water because of the accident, raising the core temperature to 1,200 degrees Celsius on March 12. Nuclear fuel pellets and cladding pipes melted down due to high temperatures, and a 20 cm-thick iron reactor pressure vessel melted down. As the zirconium, which is the cladding material of nuclear fuel, exceeded 1,200°C, a hydrogen explosion occurred at Unit 1 on the 12th, Unit 3 on the 14th, and the 4th on the 15th, damaging the containment vessel, and the radioactive materials were released to the atmosphere began. The accident marked the highest level of the International Nuclear Accident Class (INES), Stage 7, or Major Accident.

Japan stores radioactive contaminated water contaminated with radioactivity by rainwater and groundwater flowing under the reactor and treats it with a Multi-Nuclides Removal Equipment (ALPS). On April 13, Japanese Prime Minister Yoshihide Suga made the announcement after convening a meeting of relevant ministers to formalize plans to release the radioactive water accumulated at the plant into the ocean.

After the announcement on April 13 by the Japanese government, international organizations and countries worldwide express their opinions.

The IAEA made a statement that the IAEA is ready to provide technical assistance to Japan and Japan's ocean discharge. Director-General Rafael Mariano Grossi said, "The Japanese government's decision is a milestone that will help lay the groundwork for continuous progress in decommissioning the Fukushima Daiichi Nuclear Power Plant."

The Korean government expressed strong regret over the Japanese government's "unilateral" choice to release the radioactive water, saying the decision was made without discussions or negotiations with Korea.

Based on the timeline of key events related to the Japanese government's decision on the wastewater of Fukushima reactors and various international reactions to Japan's plans to release Fukushima water into the ocean, this paper aims to provide several considerations for developing a strategy for Japan's decision to release Fukushima Water into the ocean.

Keywords: Fukushima accident, Wastewater, Strategy

Infrastructure of State Regulation in the Field of Atomic Energy in KAZAKHSTAN

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The main goal of activities in the field of atomic energy use in the Republic of Kazakhstan is reliable and cost-effective provision of nuclear and radiation safety of present and future generations of people and protection of human environment from radioactive contamination, both in normal and extreme situations. To meet these challenges, the Government of the Republic of Kazakhstan has established and operates an infrastructure of state regulation in the field of atomic energy use, which included the State system for regulating safety of the Republic of Kazakhstan and the appropriate regulatory framework of the Republic of Kazakhstan in the field of in the field of atomic energy use.

The State Safety Regulatory System has been created to implement the most important tasks to protect the population and environment from the negative effects of ionizing radiation. It includes the relevant state bodies responsible for regulating nuclear and radiation safety, protecting public health and the environment, including handling radioactive waste, emergency preparedness and response to nuclear and radiation accidents and ensuring the physical protection of nuclear facilities, nuclear and radioactive materials and radioactive waste.

After gaining sovereignty in 1991 in the Republic of Kazakhstan, work has begun and still ongoing to create and improve the legal framework in the field of atomic energy use.

The state bodies involved in ensuring safety oversight in the use of atomic energy have developed and put into effect regulatory documents within their competence.

Depending on the level and degree of coverage of issues and scope, technical regulations, rules and standards in the field of atomic energy use in the Republic of Kazakhstan are enacted by the Decree of the Government of the Republic of Kazakhstan or by an order of the relevant ministry.

Radiation situation on the territory of the Republic of Kazakhstan is currently determined by the main factors (activities of enterprises of uranium mining and processing industry and related geological works, activities of nuclear fuel cycle enterprises, including producing nuclear fuel for nuclear power plants and research reactors, activities of mining and processing enterprises, whose raw materials are characterized by a high content of radioactive elements, consequences of nuclear explosions conducted for the military and peaceful purposes at test sites located on the territory of Kazakhstan, energy and research nuclear installations, using of radioisotope products in industry, medicine, agriculture, science and research).

Existing infrastructure allows for effective regulation in all designated areas of atomic energy use in Kazakhstan during over 25 years, and work is constantly being done to improve it.

Keywords: Regulation, Infrastructure, KAZAKHSTAN, Safety

Societal and Environmental Impact Analysis of Different Nuclear Fuel Cycle Scenarios

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The economic analysis of different fuel cycle options has been carried out for looking for the viability of the recycling nuclear fuel cycle option. The other factors such as energy security and social acceptability, which are mostly difficult to convert them to cost, have played an important role to the decision making. In this respects, this paper is trying to compare the social and environmental impact in once-through option and pyroprocessing recycling option (Pyro-SFR option) for giving more information to the decision makers.

In this study, the evaluation indicators are induced from a recent OECD/NEA report, “Strategies and Considerations for the Back End of the Fuel Cycle”. A total 10 evaluation indicators with three groups were selected for the comparison of the two fuel cycle options. Three groups consist of “challenges to development of the fuel cycle option” such as technical challenge, “opportunities when fuel cycle option is implemented” such as flexibility towards energy independence, and “risks when fuel cycle option is implemented” including public and environmental safety.

It is resulted from the evaluation of social and environmental impact that direct disposal option is superior to Pyro-SFR option in terms of technical challenge, and both options are similar each other in view of social acceptance and worker safety. It is, however, indicated that Pyro-SFR option is superior to direct disposal option for other seven indicators. For example, it is considered that the PWR Pyro-SFR system has more opportunities to provide direct and indirect benefits for economic development, such as building additional facilities, creating new jobs, and revitalizing the local economy, compared to the once-through cycle. In terms of preserving natural resources and energy independence, the Pyro-SFR system is more preferable than the once-through system because it recycles spent fuels which were discharged from PWRs for a fuel for SFRs.

Overall, it would be resulted for the evaluation that Pyro-SFR option has many benefits for the social and environmental impact and accordingly it should be considered in decision making of the fuel cycle options when performing the analysis of the economic feasibility.

Keywords: Societal impact, Environmental Impact, Nuclear Fuel Cycle Options, Pyroprocessing

2분과

사용후핵연료 처분전관리 (Oral)

OASIS Canister Models of the Damaged Spent Fuel Storage for Kori-1,2

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Two OASIS canister models, OASIS-37KD and OASIS-32KD, have been developed for the storage of the damaged spent nuclear fuel of Kori-1,2. These damaged fuel storage canister models are derived from the OASIS-37K and OASIS-32K canister models which can load 37 intact fuel assemblies by applying burnup credit and 32 intact fuel assemblies of unburned fresh condition, respectively. The design parameters of the damaged fuel storage canisters are identical to those of the intact fuel storage canisters except the damaged fuel storage cell design. The cell for the damaged fuel assembly is designed to have large dimension compared to the intact fuel cell to accommodate the damaged fuel storage can which encases the damaged fuel assembly. Also, the outer configuration and size of the OASIS-37KD and OASIS-32KD canisters are identical to those of previously developed OASIS canisters. Thus, the outer over-pack of the OASIS cask, i.e., OASIS-32D, OASIS-STO, and OASIS-HC, can be applied to these damaged fuel canisters for Kori-1,2 without any change.

The OASIS-37KD canister has the damaged fuel loading capacity of maximum 4 and the total fuel loading capacity of 37 including the intact fuel assemblies and the OASIS-32KD canister has the damaged fuel loading capacity of maximum 8 and the total fuel loading capacity of 32 including the intact fuel assemblies. The loading capacity of the damaged fuel assemblies of the canisters could be adjusted within the maximum capacity according to the fuel storage situation.

On the nuclear criticality safety analysis of the OASIS-37KD canister, it is assumed that the intact fuels have the required minimum burnup while the design maximum enrichment with the most reactive condition is considered on the damaged fuel. For OASIS-32KD, it is assumed that the enrichment of all fuel assemblies loaded, i.e., damaged fuels and intact fuels, is to be the design maximum enrichment with the most reactive fresh condition. The considered design maximum initial enrichment of the fuel assemblies loaded in the canister is to be 4.70 wt% which is higher than that of the Kori-1,2 fuel.

These OASIS canister models for Kori-1,2 allow massive loading and storage of fuel assemblies regardless of the fuel rod damage and it will provide the flexibility on the spent nuclear fuel management.

Keywords: OASIS, Damaged Fuel Canister, Spent Nuclear Fuel, Kori-1,2

Investigation of Dissolution Behavior of SrO in Molten LiCl Salt at 923 K

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In the early stage of the disposal of used nuclear fuel (UNF), ^{90}Sr and ^{137}Cs are the main contributors to decay heat and radiation from UNF. Therefore, the separation of these fission products from UNF can reduce the heat generation and increase space utilization of the disposal repository of UNF. Pyroprocessing has advantages in the separation of alkali and alkaline earth metal oxides because most of the group of metal oxides spontaneously dissolve in molten salt, in particular in LiCl. However, the dissolution mechanism of these metal oxides and their solubility were not fully understood yet. In this paper, the dissolution behavior of SrO, which has been known as the least soluble among other salt-soluble fission products (Cs_2O, BaO, and Rb_2O), in molten LiCl salt at 923 K was investigated. The concentration of SrO in LiCl and the dissolution products were analyzed using Inductively Coupled Plasma Optical Emission Spectroscopy (ICP-OES) and X-Ray Diffraction (XRD), respectively, after the dissolution of SrO pellets in LiCl. In addition, the melting points of the samples were measured using Differential Scanning Calorimetry. The concentration of SrO in LiCl was measured up to 50.79 wt%, and Sr_4OCl_6 was observed as the dissolution products. The melting points decreased as more SrO dissolved in LiCl, and the eutectic melting point was estimated near at 21 mol% of SrO. Finally, the equilibrium of the dissolution reaction derived from the experimental results was compared with the simulated results using HSC chemistry 10. This finding could be used to reduce the toxicity of UNF and enhance the safety of the disposal repository of UNF.

Keywords: Used nuclear fuel, Pyroprocessing, Highly heat-generating fission products

Studies on the Chlorination Reaction of SrO and BaO in LiCl-KCl Molten Salt

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This paper demonstrates the chlorination of SrO included in the simulated LWR fuel using LiCl-KCl molten salt to separate one of the major high-decay-heat dissipating nuclides, strontium, for reducing the disposal area of spent nuclear fuel.

In the first experiment, an anhydrous LiCl-KCl eutectic salt was used at 500°C in the presence of various concentrations of SrO to derive the chlorination reaction of the oxides with LiCl. As confirmed through the thermodynamic calculations using the HSC chemistry, SrO and BaO were completely dissolved (chlorinated) in LiCl-KCl salt within about 1 hr up to the 3-fold higher concentrations of the respective oxide than the expected concentration calculated based on the burn-up, cooling time, and initial U enrichment of the fuel.

Second, the simulated fuel containing SrO and BaO was fabricated in pellet and powdered forms to chlorinate them using LiCl in 500°C LiCl-KCl salts for 1 hr. The powder-type simulated fuel was prepared by a thermal heat treatment of the simulated fuel pellet at 500°C for 1 hr under an air atmosphere. The particles size was observed to be in the range from about 5 μm to 20 μm by SEM. After the chlorination reaction, the bulk salt from each crucible containing LiCl-KCl salt with a different fuel was sampled for chemical analyses. In the ICP-MS results, The chlorination of SrO was found to be more effective in the powder-type fuel compared to the pellet-type one, while a difference of chlorination reaction kinetics of BaO between the two types of the fuel was relatively smaller than that of SrO. This shows that a certain portion of SrO exists in the U matrix as a solid solution with uranium oxide, while BaO is known to exist in the grain boundaries, which is relatively easy to contact with molten salt, facilitating the chlorination reaction with LiCl.

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

Simulation of Spent Fuel Behavior of PWR for Different Burnups via FRAPCON-4.0 Modification

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From a safety point of view, it is important to determine the various conditions of spent fuel after dry storage. The behavior of spent fuel is largely influenced by the condition during operation. Therefore, the spent fuel should not be considered independently, but in a continuous state from steady-state operation. In a similar vein, as the degree of burnup varies, the state of the nuclear fuel is all different during operation, and the behavior of the spent fuel after dry storage is also different.

In this study, several parts of FRAPCON-4.0 were modified and used so that they can be applied to spent nuclear fuel. First, when dry storage starts, it is set to suppress oxidation and hydrogen pickup. Next, during storage period, the pellet swelling behavior continues to occur. Therefore, the swelling rate was newly calculated using the best-estimate equation of spent fuel. Similarly, for spent fuel, a new calculation must be made by applying the appropriate creep rate equation. Among the dating creep models, the Grain Boundary Sliding equation was used. Finally, since there is no coolant during dry storage, FRAPCON-4.0 sets the temperature of all nodes equally to the inlet temperature. However, in reality, even during dry storage, the temperature distribution in the axial direction exists. Therefore, the temperature distribution in the dry storage period was added.

This study focused on PWR nuclear fuel, operated for 54 months with burnup of 60 MWd/kgU. The behavior of spent fuel after 5 years of wet storage, and 100 years of dry storage is investigated. In addition, the discharge burnup varied in the range of 40~90 MWd/kgU and the behavior of the spent nuclear fuel is compared.

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Keywords: Dry storage, Spent fuel, FRAPCON, PWR

Characteristics of Multiaxial Stress-induced Hydride Reorientation in Zirconium Alloy Cladding Tubes

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When the excessive hydrogen absorbed to Zircaloy cladding tubes as a result of oxidation precipitates under hoop stresses above a threshold, radial hydrides in the cladding matrix are formed during cool-down in dry storage. The radial hydrides deteriorate the mechanical integrity of cladding.

Recent studies have demonstrated that the hydride reorientation under multi-axial stresses occurs at a threshold stress lower than that of the uniaxial loading. As an extension of the multi-stress effect, this study experimentally investigates the effect multi-axial stress on hydride reorientation of pressurized Zircaloy cladding tubes for a range of hydrogen contents. The observed multi-axial stress effect on the tubular cladding was analyzed by applying a modified thermodynamic model for hydride reorientation under multi-axial stresses. The crystallographic orientations of the hydride-matrix interface employed in the thermodynamic model was attained by Electron Backscatter Diffraction (EBSD) analyses.

The regulatory implications of the multi-stress effect on the spent fuel management will be discussed in the end.

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Keywords: Hydride reorientation, Multi-axial stress, Dry storage, EBSD, Microstructure

An Isotopic Analysis of SFCOMPO Radiochemical Assay Data Using TRITON and ORIGEN-S Calculation Results for Evaluating Uncertainties in Burnup Credit Application

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Criticality safety evaluation of spent nuclear fuel employing burnup credit requires evaluating the uncertainties in compositions of actinides and fission products from irradiation in a reactor. The evaluation of the propagation of the isotopic uncertainties to criticality calculation is usually conducted in two steps. In the first step, the bias and bias uncertainty for each nuclide were determined by comparing the measured compositions of the radiochemical assay data and the ones calculated using the burnup calculation code. In the second step, the criticality calculation is performed for a specific spent fuel storage facility considering the uncertainties from the first step to obtain the bias and bias uncertainty for k_{eff}. One of the popular methods used in the second step is the Monte Carlo sampling method. The isotopic compositions are randomly sampled with the probability density distribution for the ratio of the measured-to-calculated isotopic ratios, and they are used for the criticality calculations.

The final objective of our work is to evaluate the propagation of the isotopic uncertainties to k_{eff} for our domestic PWR spent fuel storage pool. In this work, the appropriate samples from the international Spent Fuel Isotopic Composition Database (SFCOMPO) were selected, and they are used to evaluate the isotopic uncertainties using TRITON and ORIGEN-S coupled with the ENDF/B-VII.r1-based 252 group cross section library. We selected nineteen experimental data of Ohi-1, Ohi-2, and Turkey Point-3 PWRs because they have similar features to the fuel assemblies of a domestic PWR with a thermal output of 950 MW, and they give the detailed operational history parameters that are important for burnup calculation (e.g., burnup, specific power, initial uranium enrichment, fuel rod specifications, critical boron concentration, irradiation cycle, etc.). In particular, we considered the irradiation history of each sample, and the inventories of actinides and fission products were calculated through two-dimensional modeling of the fuel assembly with TRITON. Also, the ORIGEN libraries were produced with TRITON, and the problem-dependent burnup calculations were performed through the ORIGEN-ARP module using multidimensional interpolation methods. These isotopic calculation results are compared with experimental data.

The validation results show good agreement for the actinides relatively to the fission products. Additionally, better agreements for the uranium isotopes were shown relatively to the plutonium isotopes. Specifically, both TRITON and ORIGEN calculations show the differences of less than 5% in uranium isotopes and 10% in the case of plutonium isotopes compared to the measured values. But Pu-242 shows a difference of more than 15% depending on the samples. Exceptionally, the Ohi-1 sample showed a difference of 14% for U-235, but the previous studies also showed a difference of 14%. Also, it was shown that TRITON calculation results were closer to the measured data than ORIGEN calculation, but the differences between the two codes were not significant.

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Keywords: Isotopic composition predictions, Depletion uncertainty, Burnup credit, SCALE, SFCOMPO

Data Analysis of Surrogate Spent Nuclear Fuel Loaded Road Transportation Test Under Normal Conditions of Transport

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Road transportation tests under normal conditions of transport using surrogate spent nuclear fuel were performed in September, 2020 using a test model of KORAD-21 transportation cask. In order to investigate amplification or attenuation characteristics, according to the load transfer path, thirty-five accelerometers were attached on the trailer platform, cradle, cask, canister, disk assembly, basket, and surrogate fuel assemblies. To investigate the durability of spent nuclear fuel rods, thirty-two strain gages were attached on surrogate fuel assemblies. Handling test and Transportation test were conducted and the transportation test were composed of five different test conditions: speed bump, lane change, deceleration, obstacle avoidance and pilot test. All tests were conducted in Doosan Heavy Industries and Construction. Data analysis of the test was conducted along with detailed review in the aspect of load transfer characteristics and durability. Amplification or attenuation characteristics, according to the load transfer path were investigated. As a result, when the load was transferred from the trailer to the cask, the load was attenuated more than ten times. On the contrary, when the load was transferred from the cask to the surrogate spent nuclear fuel assembly, it was amplified by two to three times. Representative shock response spectrum and power spectral density of several test modes were also obtained. Based on the strain data obtained from the test results, an imaginary road transportation scenario was established and based on the scenario, a fatigue evaluation of spent nuclear fuel rod was performed. As a result of the evaluation, no fatigue damage occurred on the fuel rods.

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Keywords: Normal conditions of Transport, Spent Nuclear Fuel, Structural Integrity

2분과

사용후핵연료 처분전관리 (Poster)

Comparison of Hydride Embrittlement of Zircaloy-4 and Zr-Nb Alloy Cladding Tubes

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In the spent fuel storage phase, nuclear fuel cladding is subjected to increased embrittlement due to a large amount of hydride precipitation. This study investigates the effect of hydrogen-induced cladding embrittlement of Zircaloy–4 and Zr-Nb alloy cladding tubes with ring compression and ring tensile tests at the temperature of spent fuel pool ~40°C. Tested claddings are selected as a reactor-grade stress relief annealed (SRA) tubular cladding currently used in a commercial nuclear power plant in order to derive direct implications for spent fuel management by analyzing experiment results. As a result of the experiment, an abrupt ductile-to-brittle (DTB) transition was found at hydrogen content of ~600 wppm in Zircaloy-4 cladding, and ~500 wppm in Zr-Nb alloy cladding.

To explain the abrupt ductility loss of Zircaloy-4 and Zr-Nb alloys, hydride interlinks thermodynamic model was introduced. The model explains that when Zr alloys are hydrided to some significant extent, the hydrides tend to connect to one another and form an interlinked structure. The difference in the critical hydrogen concentration for the DTB transition of Zircaloy-4 and Zr-Nb alloys are compared. According to hydride interlinks thermodynamic model, this difference appears to be originated from the grain boundary (GB) structural connectivity resulting from the difference in the grain size of the material.

Electron Backscatter Diffraction (EBSD) is used in measuring grain size difference and analyzing the crystallographic orientation of the matrix–hydride interface in these two materials. EBSD analysis shows that Zr-Nb alloy has a relatively small grain size than Zircaloy-4. Since GB is the preferred site for hydride nucleation and the interconnected hydride network is more likely to be formed, the critical hydrogen amount of Zr-Nb alloys for the abrupt ductile-to-brittle transition is lower than that of Zircaloy-4, implying reduced hydride embrittlement resistance.

From the viewpoint of spent nuclear fuel management, abrupt hydride embrittlement through DTB transition should be considered for high discharge burnup spent fuel, and the grain size control of the spent nuclear cladding via heat treatment during the vacuum drying can be suggested as a plausible engineering strategy to mitigate hydride embrittlement.

Acknowledgements

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Keywords: Zr alloy, Hydride embrittlement, Ductile-to-brittle transition, Spent nuclear fuel management

Facilitate Measurement of RHF by Using PROPHET and Its Application

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Hydrogen embrittles in the cladding of nuclear fuel rods that have been operated for a long time in a nuclear reactor. The embrittlement of hydrogen have an effect on lowering the ductility during post-treatment. At present, it is evaluated that the problems related to hydrogen embrittlement during wet storage and rearrangement during dry storage afterwards contribute greatly to the deterioration of the ductility of the cladding. At the time when the necessity of dry storage is emerging, many studies related to rearrangement of hydrides are being conducted.

This hydride rearrangement phenomenon can be easily observed with an optical microscope. For this, codes that can measure the hydride rearrangement ratio (RHF) are being developed in several places. In this study, I would like to introduce an accessible code that can easily measure the RHF if the picture quality is visible with the identification code called PROPHET. It is possible to easily grasp the RHF value of the desired part by receiving the region of interest. Through this, we will find out the effectiveness of hydride recognition by comparing the concentration and the recognized hydride lengths. And, based on this applicability, we will find out a direct comparison of the RHF values according to the stress distribution. Finally, the efficiency of the RHF value will be investigated by comparing the measured RHF value with the fracture behavior.

Acknowledgements

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Keywords: Nuclear, Hydride Reorientation, Hydride Reorientation Fraction, RHF

Hydrogen Diffusion in Zirconium Alloy During Dry Storage by Simulation and Experiment

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At the end of operation of light water reactor, zirconium alloy contains hydrogen solid solutions which absorbed by waterside corrosion. Hydrogen changes phase to hydride depends on temperature and concentration. At the beginning of the dry storage, fuel rod is heated up to 400℃. The high temperature and temperature gradient in dry cask cause hydride to dissolute into solid solution and diffuse by Fick's law and Soret effect. The diffused hydrogen is concentrated at the cold end by Soret effect. Consequently, the concentrated hydrogen become to hydrides along terminal solid solution curve and cause brittleness to the material. The present work predicts hydrogen concentration in the ZIRLO tube through 1D diffusion and phase transition simulation and verifies the calculation through experimentation. This research improved the prediction of hydrogen concentration by modifying concentrating effect at the cold end in simulation. Additionally, the experiment result is compared and analyzed with simulated result.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation O f Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security C ommission (NSSC) of the Republic of Korea (No. 2003018).

Keywords: Hydrogen diffusion, Hydride, Zr alloy, Dry Storage

Vibration Characteristics of KJRR-F Fresh Fuel Under Normal Transportation Test

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KJRR-F PSAR "Permitted accelerations to handle and transport of nuclear fuel assembly" stipulates that the maximum acceleration applied to the KJRR-F fresh fuel assembly should not exceed six times the gravitational acceleration (6 g) under the normal transportation conditions. In order to check that the KJRR-F transportation cask and its tie-downs structure complies with the above regulation (6 g under normal transportation conditions), the normal transportation test from KAERI (Manufacturing site) to Kijang (Reactor site) was conducted to evaluate the structural integrity of KJRR-F fresh fuel. From the test result, the maximum acceleration at the fuel assembly was 4.57 g at the test which had the fastest average speed (80 km/h). Additionally, through the normal transportation test from KAERI to Kijang, the vibration characteristics of the KJRR-F fresh fuel with its transportation cask and tie-down structure was identified. Since the maximum acceleration (4.57 g) is lower than 6 g as specified in the KJRR-F PSAR, the test presented that there is no problem in transporting the KJRR-F fresh fuel using the KJRR-F transportation cask and its tie-down structure.

Keywords: Vibration characteristics, Normal transportation test, KJRR-F fresh fuel

Development of PHWR Spent Fuel Transport Basket

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HI-STAR63 transport cask are used to transport PHWR spent nuclear fuel from a wet storage pool to a dry storage facility in KOREA. In particular, it is equipped with two cylinder-shaped baskets containing 60 bundles of PHWR spent fuel in HI-STAR63 transport cask. However, the existing basket is designed as airtight container for dry storage with very low radiation shielding performance and must be cut to get fuel out of the basket by welding sealing. Therefore, this study developed transport baskets that are easy to transport and handle at KAERI hot cell facility, to prepare the hot cell test required to verify long-term integrity of PHWR spent nuclear fuel. HI-STAR63 transport cask will be used for offsite transport of PHWR spent nuclear fuel, and the developed transport basket will be installed inside HI-STAR63. The transport basket was designed to ensure structural, radiation shielding, containment, and thermal safety, and finally demonstrated its performance after manufacturing is complete. It will be used for transportation to a hot cell facility to verify integrity of PHWR spent nuclear fuel under long-term storage.

Keywords: Spent Fuel, Transport Basket, PHWR, HI-STAR63

Drop Condition and Sensor Position Determination for Performance Test of KTC-360 (High Capacity Cask for CANDU Spent Fuel Bundle)

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A high-capacity cask for transporting spent nuclear fuel from heavy water reactors named KTC-360 has been developed by KONES, Moojin Co. and KAERI. The high-capacity cask is capable of transporting a total of 360 bundles of CANDU spent nuclear fuel bundles. This transport cask is capable of carrying 6 baskets loaded with 60 bundles of spent nuclear fuel in each basket.

For the structural performance validation test of this transport cask, the drop cases were determined and the sensor locations were selected. The structural test consists of 5 drop cases and 2 puncture cases. Through finite element analysis, drop postures with a large acceleration value were selected as test cases. The selected drop test postures are Bottom drop, Lid drop, Long side drop, Long lid edge drop, and Long side oblique drop. The puncture test was selected in the posture that could bring the greatest damage to the containment boundary of the cask. The selected puncture test postures are Side puncture and Lid puncture.

The structural test model will be produced as a 1/2 scale test model. A total of 30 strain gauges are installed in the structural test model. The strain gauges are installed in 4 locations on the lid of the cask, 4 on the outside of the side of the cask, and 4 on the bottom of the cask. A total of 10 strain gauges are attached to the inner cylinder inside the cask. In addition, to evaluate the axial force of the bolt, a bolt sensor is manufactured and installed in a total of 8 locations, each of 4 in each inner cylinder. The bolt sensor is installed in 4 locations in the direction of 0, 90, 180 and 270 degrees based on the lid of one inner cylinder. When the sensor is installed inside the cask, it is necessary to route the sensor cable so that it can be drawn out. For this, the perforated hole should be fabricated on the lid. For the cable routing of the sensor attached to the bottom of the cask, a groove was fabricated on the side of the cask so that the cable could be pull out without damage. In order to maintain the drop posture during the drop test, three additional lugs are attached to the side of the cask for ease of drop test though they are not in the original cask. The structural safety test and evaluation is scheduled to be conducted in the second half of this year.

Acknowledgements

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Keywords: High Capacity Cask, CANDU Spent Nuclear Fuel, Drop Test Posture, Sensor Location

Evaluation of Fuel Depletion Uncertainty in Criticality Analysis of GBC-32 Cask

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The interim storage capacity of domestic PWR used nuclear fuels (UNFs) will reach the limit from 2029. It is necessary to utilize a dry storage cask system to store UNFs and hence the accurate criticality analysis of the cask is very important. A criticality analysis with burnup credit is performed as follows: (1) the calculation of isotopic compositions within a UNF to be stored into a cask by a depletion assessment code and (2) the calculation of the multiplication factor (k_{eff}) value for a cask by a criticality assessment code. In particular, the isotopic compositions by a depletion analysis should be accurately estimated since they strongly affect the accuracy of criticality analysis. So, it requires depletion uncertainty in k_{eff} by a depletion code for a criticality analysis with burnup credit. The objective of this work is to assess the depletion uncertainty in k_{eff} for the generic 32 PWR-assembly burnup credit (GBC-32) cask resulted from isotopic biases and bias uncertainties in the nuclide concentrations for PLUS7 UNFs. First, the bias and bias uncertainty in k_{eff} for the GBC-32 cask system were assessed using the Monte Carlo uncertainty sampling method for three different burnups (i.e., 10,000, 30,000, and 50,000 MWD/MTU) and the isotopic biases and uncertainties obtained with the calculated and measured isotopic concentrations. The PLUS7 UNFs discharged from the Hanbit Nuclear Power Plant Unit 3 were assumed to be loaded into the GBC-32 cask. The biases and bias uncertainties in keff were calculated for three specific burnups using the SCALE 6.1/STARBUCS code. Second, the statistical F-test in the two-way analysis of variance (ANOVA) was suggested to predict the statistically converged total depletion uncertainty in k_{eff}, with a reduction of computing efforts in the Monte Carlo criticality calculations. Finally, the statistical F-tests in the two-way ANOVA were utilized to efficiently assess the value of a reasonable total depletion uncertainty in k_{eff}. As a result, 100 criticality calculations were needed to get a reasonable total depletion uncertainty for 10,000 and 30,000 MWD/MTU, while 120 criticality calculations were needed for 50,000 GWD/MTU. With these methods, the reasonable total depletion uncertainties in k_{eff} for 10,000, 30,000, and 50,000 GWD/MTU were efficiently estimated to be 0.011866, 0.013175, and 0.030811 Δk, respectively.

Keywords: Dry storage cask, Criticality analysis, Depletion uncertainty, Monte Carlo uncertainty sampling method, Statistical F-test

Review on Application of Spent Nuclear Fuel Dry Storage Canisters

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Spent nuclear fuel dry storage system is classified into a canister-based system and a cask-based (*or* bare-fuel) system depending on whether or not the canister is used for loading and storing fuel regardless of the type of dry storage system. The canister-based system requires welding the upper part of canister after fuel loading and moving the canister to the storage facility with a separate transfer cask (*or* transport cask), so that the operating process of canister-based system is more complicated than the cask-based system. In particular, the United States has applied the canister-based systems to most dry storage systems and has planned to dispose of the fuel-loaded canisters without repackaging them as well as for transport and storage at nuclear power plant and interim storage facilities. These canister-based systems were evaluated for safety in compliance with the regulatory requirements and licensed for transport and storage, but no licenses were not acquired for disposal. The safety design and regulatory requirements are the same for the cask-based system and the canister-based system. Spent nuclear fuel dry storage systems using the canisters could provide linkages between various SNF management steps of nuclear power plant storage, interim storage and disposal. The canister-based systems reduce the amount of fuel handling and thereby reduce radiation exposure. The canister-based systems have the potential to simplify the operations of connected storage and disposal facilities, which make achievement of safety easier, and less costly. The canister-based systems for transport, storage and disposal must meet the long-term requirements of disposal, and include features to ensure subcriticality conditions, perform as engineering barrier to ensure substantially complete containment and meet thermal requirements for many thousands of years in the disposal environment. This study reviewed the use of canister-based dry storage systems from nuclear power plants, their use for interim storage and the possibility of canister direct disposal.

Keywords: Dry storage system, Canister-based system, Cask-based system, Interim storage, Disposal

Handling Analysis for SAFER Under Abnormal Installation Conditions

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A lot of Spent Nuclear Fuels (SNFs) have been stored in SNF pools of the each pressurized water reactor nuclear power plant. The top nozzles of some SNFs were separated during handling operation due to the Inter-Granular Stress Corrosion Cracking (IGSCC) and/or Irradiation Assisted Stress Corrosion Cracking (IASCC) due to sensitization, neutron irradiation and others. Therefore, handling safeties of the SNFs with IGSCC and/or IASCC are required to be enhanced.

To prevent this potential problem, SAFER (Safely Adapting FastenER), which is a handling reinforcement device, was developed by KEPCO Nuclear Fuel. The SAFERs are installed at bulge region in guide tube of SNF, and that was designed to use 4 SAFERs per SNF under normal condition. However, 1 or 2 SAFER(s) cannot be installed due to unexpected states, environments and conditions in SNF. To analyze behavior response of SNF under these abnormal installation conditions, non-linear dynamic Finite Element (FE) analyses were performed on unsymmetrical (3 SAFERs) and diagonal (2 SAFERs) installation condition for 14OFA fuel, respectively.

The components of FE model are a fuel handling tool, a top nozzle and 2 (or 3) insert-guide tube assemblies. The rest SNF mass without the generated fuel components and the added mass of SNF due to handling in water are adjusted using a mass element, which is located at center of SNF: the added mass is calculated from water mass of square column volume considering conservative dimensions of SNF. The guide tubes and mass element are connected by rigid elements. The installed SAFER is simulated as rigid element, which are connected between top surface of top nozzle and bulge in guide tube. Loading condition is a lifting velocity of the tool, which is based on the current crane working velocity in the power plant. The velocity is applied at the center point on top surface of the tool, and the 6 degrees of freedom on the point are not constrained to simulate the hanging of the tool on the crane.

The axial alignment values of SNF under abnormal installation conditions are obtained from the analyses. The non-dimensional alignment values, which mean distances in the cross-section plane between centers of top and bottom of SNF, are 1.11 (3 SAFERs) and 0.15 (2 SAFERs), respectively. These values are significantly less than the non-dimensional length (7814.23) of the fuel: all non-dimensional values are calculated based on the tolerance of envelop of 14OFA. Therefore, normal handling operation can be possible, although SNF is under abnormal SAFER installation condition.

Keywords: Spent nuclear fuel (SNF), Handling reinforcement device, SAFER (Safely adapting fastener), Abnormal installation condition, Non-linear dynamic FE analysis

Evaluation of Residual Water After Drainage in the Canister

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As the spent pool storing spent nuclear fuel (SNF) is expected to be saturated, a method of dry storage of SNF has emerged. The series of work processes that transport SNF from the spent pool to the dry storage facility is called the short term operation process. The short term operation process involves loading the cask with fuel, moving the cask, draining the cask, drying and filling with inert gas. In this paper, we evaluated the water remaining after draining the water in the canister during the short term operation process.

The water inside the canister is drained using a mechanical pump. Inside the canister, physisorbed water remains on the surface due to surface tension even after water is drained, and residual water may exist in the interior of the canister, SNF and base plate. The evaluation test model for residual water was selected as KORAD-21 and calculated using the solidworks program. First, the KORAD-21 canister was 3D modeled with solidworks program to calculate the internal volume and surface area. According to the related literature, it is stated that the concentration of moisture on all external surfaces of SNF (e.g. cladding pipes, nuclear fuel bundles) and canister interior materials (inner walls, baskets, etc.) is present in the surface layer from 0.03 g/m^2 to 0.05 g/m^2. Residual water may exist in the canister base plate and spent nuclear fuel as well as physisorbed water. This residual water was calculated by calculating the volume of the drain hole of the canister base plate, and was calculated based on the results obtained through the test of the model ACE 7 spent nuclear fuel. As a result of the calculation, it is predicted that a total residual water of 0.443 L to 0.449 L will remain inside the canister.

By performing a drying process to remove residual water through residual water evaluation, it is possible to secure the safety of spent nuclear fuel and internal structures during long-term storage of spent nuclear fuel.

Keywords: Spent Nuclear Fuel, Dry Storage, Canister, Residual Water, Short Term Operation

Preparation of Ag-Containing Aluminosilicate Sorbents With Improved Textural Properties

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Ag-containing hydrophilic aluminosilicate aerogels capable of capturing iodine were prepared by sol-gelation from Na, Al, and Si alkoxides using co-solvent exchange, Ag/Na ion exchange, solvent exchange, and ambient-pressure drying. The Na^+AlSi-OH gel was prepared using sodium methoxide (NaOMe): aluminum tri-*sec*-butoxide ($Al(O^sBu)^3$): tetraethyl orthosilicate (TEOS) molar ratios of 1.05:1:1 and 1.3:1.1:1. For solvent exchange step, excessive $AgNO_3$ in pores of Ag^+AlSi-OH hydrogels after Ag/Na exchange was exchanged with purified water and then water in those was exchanged by isopropanol. Finally, isopropanol in pores was exchanged using n-heptane. In this work, the solvent effect on textural properties such as Brunauer-Emmett-Teller (BET) surface areas and pore size distributions was investigated using water (72.7×10^{-3} N/m) with high surface tension, isopropanol (21.7×10^{-3} N/m) and n-heptane (20.14×10^{-3} N/m) with low surface tension.

The textural properties of Ag^oAlSi-OH sorbents were significantly improved with a decrease in surface tension of solvent. As surface tension of solvents decreased, the BET surface area of sorbents was increased from 125 to 174-179 m^2/g. The water-exchanged sorbents showed the pore size distribution in the mesoporous range from 4 to 40 nm, whereas the both isopropanol- and n-heptane-exchanged sorbents had that in the mesoporous and macroporous ranges of 4-107 nm and 4-133 nm, respectively. Thus, the isopropanol- and n-heptane-exchanged sorbents exhibited higher specific pore volume and average pore diameter than the water-exchanged sorbents. The cumulative pore volume and average pore diameter of isopropanol- and n-heptane-exchanged sorbents were 1.08 cm^3/g and 20.8 nm, 1.29 cm^3/g and 24.2 nm, respectively.

All solvent exchanged sorbents had pore diameters much larger than the kinematic diameter of I_2 (0.5 nm) and HI (0.35 nm). Hence, almost all I_2 and HI in the off-gas stream could easily enter the pores of sorbents and be transported to Ag nanoparticles through these large pores. The large pore diameter and higher pore volume than AgX (0.28 cm^3/g) and Ag^0Z (0.1 cm^3/g) will be expected to result in a high degree of Ag utilization.

The n-heptane-exchanged sorbents among solvent-exchanged sorbents exhibited the best characteristics with a maximum iodine capture capacity of 0.613 g-I/g-sorbent, a BET surface area of 179 m^2/g, a cumulative pore volume of 1.29 cm^3/g, and an average pore diameter of 24.2 nm.

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Keywords: Silver, Iodine capture, Hhydrophilic sorbent, Solvent effect

Preparation for 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 under normal conditions of transport using surrogate spent nuclear fuel were performed in September, 2020 using a test model of KORAD-21 transportation cask and sea transportation tests is planned in September, 2021. In order to investigate amplification or attenuation characteristics, according to the load transfer path, a number of accelerometers will be attached on a ship cargo hold, cradle, cask, canister, disk assembly, basket, and surrogate fuel assemblies and to investigate the durability of spent nuclear fuel rods, thirty-two strain gages will be attached on surrogate fuel assemblies. The strain gage attachment locations will be same with those of the road transportation tests. A ship which has similar deadweight of existing ships for transportation of spent nuclear fuel will be rented for the sea transportation tests. To investigate the vibration characteristics according to the loading position in the cargo hold, four or five IMU (Inertial Measurement Unit) sensors will be attached to the different locations in the cargo hold. Weather information such as significant wave height, wind speed and wind direction are needed to analyze the test data but it is very difficult to measure the wave height on the moving ship. The Korea Meteorological Administration provides weather data including the wave height, wind speed and wind direction measured from weather buoys. Those data will be used for the analysis of the sea transportation tests.

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Keywords: Normal conditions of Transport, Spent Nuclear Fuel, Sea Transportation, and Structural Integrity

Analysis of the Effect of Various Factors on the Spent Nuclear Fuel Transportation Risk Assessment

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The spent nuclear fuel released from the light water reactor is stored in spent fuel pool in the power plant. As of the third quarter of 2020, the amount of domestic spent nuclear fuel stored is 19,887 assemblies. This amounts to 74.2% of the total storage capacity of spent fuel pool. Considering the current amount of spent nuclear fuel generated, there may be a situation in which operation is impossible due to the saturated spent fuel pool in the plant even though the operation period of the power plant remains sufficient. Therefore, as an alternative to this, research on dry storage and disposal facilities has been actively conducted in recent years.

Thorough preparation and various technologies are required to safely manage spent nuclear fuel. The spent fuel transportation risk assessment is one of the important activities related to safety. For the risk assessment, structural and thermal evaluation of the transport cask for the accident conditions should be preceded. After that, radiological consequences and potential risks evaluation is performed using codes such as RADTRAN and INTERTRAN.

In this study, RADTRAN code was used to review various factors for spent fuel transportation risk assessment and the effect of each factor on the exposure dose was analyzed. As a result of analyzing the effect of each factor on the exposure dose, the distance between the driver and the transport cask had the greatest effect on the driver's exposure, and the vehicle speed had the greatest effect on the exposure on the route. In the case of exposure to the public near the route, the effect of residential shielding factors was large. In the accident condition, changes in the leakage fraction and deposition rate of particle radionuclide, and loss of shielding factors had the greatest influence.

Keywords: Spent nuclear fuel, Transportation, Risk assessment, RADTRAN

Mechanical Behavior Modeling of Spent Fuel Subjected to Vibration Load Under Land Transportation Condition

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Transportation of spent fuel is unavoidable process in current situation. According to 10 CFR 71, the integrity of spent fuel shall be maintained under normal conditions of transportation. Specifically, in shock & vibration mode, the integrity of spent fuel shall be demonstrated. In this paper, we developed the fuel rod model by referring to the INL methodology.

In order to evaluate the vibration load on the spent fuel rod, we carried out finite element analysis using ABAQUS. In this analysis, epoxy was used to simulate the bonding status between cladding-pellet interface and pellet-pellet interfaces. Cladding-pellet bonding is a typical phenomenon in high burn-up spent fuel. Fuel rod model comprised cladding, epoxy and pellets. The length of the fuel rod model was consistent with the span length between two mid grids which prop up the fuel rod. Thus, simply supported boundary conditions were applied at both ends. One end of the cladding was constrained in all translational directions and torsional rotation, and the other end was constrained in two out-of-plane translational directions. Loading conditions were applied at the center of cladding. The loading conditions are vertical acceleration data obtained from the land transportation test. The acceleration data are 12-second long and include peak amplitude part. Boundary conditions and loading conditions were focused on the center nodes of the cladding cross-section.

Fatigue damage due to vibration load is largely in the axial strain direction, because bending is the main loading mode under normal conditions of transportation. The axial strain results were shown in-phase with the loading conditions. In order to evaluate the fatigue damage, the damage fraction was introduced. The damage fraction is defined as the sum of ratios of the number of cycles to lifetime at each strain level. If the damage fraction reaches 1, failure occurs.

We will perform similar procedure to demonstrate the integrity of spent fuel under marine transportation condition.

Keywords: Spent fuel, Modeling, Vibration load, Land transportation

Speculation on the Technical Approaches Based on Advice of Spent Fuel Public Discussion Reexamination Committee

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As the operating hours of the nuclear power plants accumulate, the amount of spent nuclear fuel (SF) has been increasing. SF rods are temporarily stored in each nuclear power plant because the disposal method of the SF has not been determined yet. However, as SF storage facilities are becoming saturated, the SF managing technique is being more critical to handle SF more safely. Based on the advice recently published by Spent Fuel Public Discussion Reexamination Committee (SF Public Discussion RC), it can be seen that the retrievability of SF shall be considered even under the disposal stage.

As of now, SF integrity evaluation is considered until only up to dry storage management level. In order to ensure retrievability in the disposal management process according to the recommendation of the Committee, reliable integrity evaluation must be carried out during the dry storage, handling and transportation stages, and related technology development shall be strengthened.

To do this end, we first need a database (DB) and evaluation model related to SF. However, since there are few DBs and models related to SF in Korea, it is essential to introduce internationally verified DBs and models and apply characteristics of domestic SF to rapidly raise the level of SF evaluation technology to the advanced level. This approach will lead our technology to the advanced level in world nuclear industries.

There are many factors that can define the characteristics of SF, but among them, the definition of degradation characteristics related to mechanical properties is the most important, and it is necessary to develop models and correlations that can predict these mechanical properties. It is very difficult to define the mechanical properties of SF, but once reliable properties are defined, the integrity evaluation can be performed very efficiently during the disposal process.

This strategy will be a shortcut to not only raise the level of SF disposal technology to an international level, but also to gain public trust.

Keywords: Spent fuel, Retrievability, Disposal, Dry storage

Design for Radioactive Waste Packages Using Honeycomb Structure as Impact Limiter

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When an accident occurs in radioactive waste transportation or storage, etc., most packages containing radioactive materials must perform a function that prevents radioactive release. Therefore, it must be designed in consideration of the load test and drop test that may occur in accident conditions.

Due to the characteristics of radioactive materials, most packages use high density metals for shielding performance, and accordingly, the shock absorption function of the package itself is very poor. So, when designing packages, an impact limiter with appropriate shape, size and material should also be chosen.

In this study, the operating scenario considers a special situation where packages are stacked on a cylinder-shaped circular tube. The static load or impact is expected to be applied in the direction of the stacking axis. For this reason, honeycomb structure specialized for axial load was designed to be attached to the bottom of the package.

The validity of the design of the package was verified using FEM (finite element method) simulation. For the dynamic analysis, Abaqus/explicit was used and elastic, plastic properties and damage criterion were applied to all materials. To assure the integrity of the package when free drop, it is compared between case of attaching the impact limiter and without it. As a quantitative perspective, absorbed energy was compared, and it also focused on whether any fracture or plastic deformation occurs in fasteners and joints. The derived results show that honeycomb structures can be a good solution for package designs that take into account static loads or impacts applied in the axial direction.

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Keywords: Package design, Drop test, Impact limiter, FEM simulation, Finite element method, Explicit method

Pre-treatment Process System for Irradiated Fuel Cladding

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To perform mechanical tests for irradiated fuel cladding, such as creep, fatigue, compression, and fatigue, the fuel cladding specimens have to be prepared adequately. After the fuel pellet is removed completely, it is essential to carry out various pre-treatment processes for the fuel cladding, such as cutting, and inspection. All pre-treatment processes have to be conducted in the radiation shielding structure to protect the operator against radiation from the fuel cladding specimens. In this research, the pre-treatment process system was newly developed. It was designed as a shielded glove box type to accommodate the fuel cladding with a fuel rod average burnup of 60 GWd/tU. The system consists of a processing cell for cutting, oxide layer removal, cleaning, and drying, and an inspection cell for defect inspection, dimension measurement, and two cells are connected by a shielded delivery tunnel. Each cell is equipped with lead glass viewing windows, shielded gloves port, and tongs enable manipulative work, and all sides are shielded with 50 mm thick pure lead, based on the shielding calculation results. The pre-treatment sequentially proceeds from the cutting of the cladding specimen. The fuel cladding is transferred from the hotcell to the processing cell via the shielded door, and then it is cut precisely into various shapes, depending on the test, using a precision cutting machine and specially designed lathe. In particular, oxide layers on both ends of the cladding, to which the high-pressure fittings are connected, are removed to prevent internal pressure leakage during the creep and fatigue test. Since defects on the inside surface of the cladding affect the mechanical test results, defect inspection for all specimens is essential. In this system, visual inspection is performed by video scope to detect internal cladding damage at the inspection cell, and then the intact cladding specimen for the test is finally determined based on the inspection results.

Keywords: Pre-treatment process, Irradiated fuel cladding, Shielded glove box

The Evaluation of Exterior Wall Radiation Shielding Performance at the PWR Spent Fuel Dry Storage Facility

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As the time of saturation of spent fuel storage pool in nuclear power plants are expected, the necessity for the establishment of dry storage facilities in nuclear power plants is increasing. Currently, the number of spent fuel dry storage facilities is increasing overseas due to the stable operation of nuclear power plants and decommissioning of nuclear power plants.

Meanwhile, Korea has developed and is operating MACSTOR, a dry storage facility for spent nuclear fuel in the Wol-seong heavy water reactor, based on the safety evaluation and design technology of the domestic spent nuclear fuel dry storage system. Based on these experiences, the development of light water reactor type dry storage facilities is actively progressing.

In this paper, the evaluation of the radiation shielding performance of the outer wall for the development of a bolt-type PWR spent fuel dry storage facility having a concept similar to that of the MACSTOR was conducted. The module, which is a dry storage facility for spent nuclear fuel in the PWR, was assumed to be a 2x5 array condition, and a multi-purpose canister (MPC)-24 type model of HI-STORM100 developed by Holtec was used for the storage canister and cylinder.

For the radiation shielding evaluation of the spent fuel storage facility, NUREG-1536 recommends that the distance to the boundary of the control area is suggested through the radiation dose rate evaluation according to the distance for the arrangement condition of at least 20 dry storage casks (typically 2x10 array).

However, in this paper, a dry storage facility with a 2x5 array condition was assumed, and the surface dose standard of the building outer wall was 0.5 mSv/hr, and the corresponding modeling was implemented to evaluate the outer wall shielding performance.

Keywords: Spent fuel, Dry storage facility, Module, Macstor, HI-STORM100

Experimental Evaluation of Heating Effect of Nuclear Fuel on Rod Surface Temperature in Forced Circulation Drying

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In order to maintain the integrity of spent nuclear fuel in a dry storage cask, water inside a canister should be drained and remaining water that could be physically bound on the surface of fuels and other components must be dried before filling inert gas such as helium. Forced—circulation drying is one of drying methods. Herein, forced-circulation drying system and canister were designed to evaluate the characteristics of forced circulation drying according to the pressure inside a canister as well as rod surface temperature depending on whether the fuel assembly is heated. A fuel assembly model manufactured was a half size of an actual fuel assembly and electric heaters were installed to simulate decay heat.

Test apparatus was composed to a canister, a helium circulation pump, a heater, a condenser and a flow controller. Three canisters with the length of 50 cm, 100 cm, 200 cm were used for drying test. After feeding 130 g of water into the canister in a desired water temperature, pressure with helium and heater temperature, forced-circulation drying test was operated until the dew point temperature reached -5°C and maintained for 9 min. Then, drying test ended up and the canister lid opened to check the residual water remaining in the canister.

In the point of the canister pressure and volume, the higher canister internal pressure and the smaller canister volume had the shorter time to the end of drying. The higher canister pressure was turned out to has the higher temperature rise of water, and it resulted in the drying time reduction. During the drying test, the drying time with the heated fuel assembly was shorter than that with the unheated fuel assembly. Moreover, the temperature change of the cladding tube should be below 65°C in ten cycles. However, in the forced circulation test, the temperature difference on the surface of the fuel assembly was about 15°C. Therefore, in the forced circulation test, the influence of the fuel assembly on the temperature limit was negligible and it is estimated that the integrity of the fuel assembly would have been maintained.

Keywords: Forced-circulation drying, Residual water, Dry storage cask, Rod surface temperature, Heated fuel assembly

Feasibility Study to Derive Design Power Spectral Density Function for Road Transportation Condition

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Through a multi-modal transportation test conducted by U.S. DOE in 2017, the United States concluded that the load imposed on the spent fuel during normal condition of transport is small, and that the integrity of the spent nuclear fuel is maintained under normal condition of transport. A similar test was conducted in September 2021 in Korea. Surrogate fuel assemblies were loaded in a full-sized transportation cask, and a road transport test was conducted within Doosan Heavy Industries & Construction. Through the evaluation of the acquired data, the same conclusions as those drawn by the U.S. DOE were drawn. It is said that the integrity of spent nuclear fuel is maintained under normal road transportation conditions. In addition, when a transport service provider transports spent nuclear fuel in the future, a different transportation cask can be used and a transport situation under different conditions may be encountered. Therefore, it will be necessary to calculate the design base load that the cask could use. The load path analysis to the fuel assembly was performed based on the load measured in the road transport test. During the test conditions such as pilot tests, speed bump tests, lane change tests, braking tests, and avoidance tests, loads covering PSD in basket and surrogate fuel assemblies were derived. Finally, a diagram envelope the power spectrum densities of the basket that transmits the load to the nuclear fuel assembly was calculated. A plan to use this as the design power spectral density function for road transportation condition was evaluated. This was compared with the PSD generated in the fuel assembly and the results were analyzed. It was found that the PSD of the pilot test had a high value in the low frequency region and the PSD of the braking test had a high value in the high frequency region.

Acknowledgements

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Keywords: Normal Condition of Transport, Road Transport Test, Spent Nuclear Fuel, Structural Integrity

Asymmetry Analysis of Mixed Composition Nuclear Materials

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It is important to characterize and quantify unknown nuclear materials in nuclear safeguards. Neutron detection based approaches have been widely used for nuclear material accounting. Recently, emission angles between prompt neutrons by nuclear fissions were investigated for developing a novel safeguards technique. The emission angle can be measured as an angular distribution, whose asymmetry is used as a parameter to analysis neutron multiplication of nuclear material samples. In previous studies, it has been confirmed that neutron multiplication could be expressed as a function of measured asymmetry values. However, asymmetry values are variable according to conditions of energy threshold applied during measurement and sample composition. Further studies are, therefore, necessary for practical applications.

In this study, the relation between asymmetry and sample composition of ^{244}Cm and ^{240}Pu, which are spontaneously fissionable nuclides, was investigated using Geant4 integrated with FREYA fission generator. The fact that angular distributions are basically averaged results of some anisotropic and isotropic portions was carefully considered. Also, relative fission rates from both nuclides were assumed by referring spent nuclear fuels of LWR, and corresponding emission angle combinations were used to define weighting factor that is related to asymmetry values. In results, linearity was confirmed in the relation between asymmetry and the weighting factor for diverse masses of samples of mixed composition. The results of this study could contribute to characterizing nuclear samples of which different spontaneously fissionable nuclides are consisting.

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Keywords: Nuclear safeguards, Nuclear material accountancy, Material characterization, Neutron detection, Monte Carlo simulation, Anisotropic emission

Effect of Structural Stiffness of a Fuel Dummy on a Drop Impact in Drop Tests of Transportation Casks for Spent Nuclear Fuels

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A drop test of transportation casks for spent nuclear fuels should be performed in accordance with the regulations. In the drop test, a simplified fuel dummy is used to simulate real assemblies of the spent nuclear fuels. The simplified fuel dummy has the same weight as the spent fuel assembly, and it is made of steel. Since the spent fuel assembly has low stiffness due to its high weight, the primary natural frequency of the spent fuel assembly is less than 10 Hz. Therefore, the simplified fuel dummy has higher stiffness than the real spent fuel assembly, which results in a much higher first-order natural frequency.

The drop impact exhibits different reaction forces depending on the stiffness of the test objects even though the test objects have same weight. Therefore, in this study, a drop analyses of the spent fuel transportation cask loaded with the simplified fuel dummy having different stiffness are performed. By calculating the load subjected to the tightening bolts for lids, the effect of the stiffness of the fuel dummy is evaluated.

Keywords: Transportation cask for spent nuclear fuel, Drop test, Drop analysis

Thermal Evaluation for Simple Model of SF* Dry Storage Module

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Thermal evaluation of the simple model was performed to set the length of time, the thickness of the insulation material, and the location and size of the air outlet through the computational fluid dynamics (CFD) analysis of the simple model in dry storage module.

The CFD analysis tool (ANSYS) was used to perform a smooth linkage analysis of pressure-velocity, and to simulate natural convection, 'Boussinesq approximation equation' was used for the working fluid (air) at a density of 30 degrees. The sensitivity was evaluated by adjusting the size of the outlet of simple model of the top center outlet type (TC, top-center), the top side outlet type (TS, top-side), and the side side outlet type (SS, side-side) in the dry storage module, and the sensitivity to heat load was evaluated.

The dry storage module used in the analysis model is set up to using a cylinder covering 10 canisters that can store 24 bundles (0.8 kW/bundle) of PWR spent fuel to block the outside air. By placing it, it was considered to satisfy the concrete temperature standard by blocking heat transfer due to convection and radiation.

The CFD analysis of simple model of the dry storage module was performed to determine whether the concrete temperature condition was satisfied or not, and the sensitivity according to the location and size of the air outlet and the cooling period was performed. It is reasonable to derive optimal temperature results through various models and apply them to basic designs in the future.

Keywords: SF (Spent Fuel), Dry storage module, Thermal evaluation, Simple model

Technical Considerations for Tomographic Verification on Spent Nuclear Fuel Prior to Transfer to Dry Storage

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Due to energy transition policies in S. Korea, aging nuclear power plants have been shut down and faced decommissioning. It consequently accelerates the saturation of the nation's entire temporary spent nuclear fuel (SNF) storage, increases the demand for building additional dry storage. Unlike on-site SNF pools, dry storage is fundamentally difficult to access and measure so SNF contents should be verified prior to transfer to dry storage. For that, several systems based on gamma and neutron tomography techniques have been developed by the IAEA and its member States. However, the problem of the poor resolution caused by the inaccuracy and quality of the measurement data is still remained.

The IAEA required to identify smaller size defects than partial defects via tomographic system, which means to verify if at least half of the fuel pins in an assembly are falsified or missing. Since the image resolution depends on the number of pixels and the size of interested defects, multiple detectors and associated acquisition electronics are required. Thus, the whole integrated system consequently becomes more complex and heavy. For instance, the PGET (partial gamma emission tomography), recently developed by the IAEA for partial and bias defect verification on spent fuel, contains 104 CdTe detectors embedded in tungsten collimators and weighs about 520 kg in air and 300 kg underwater.

Based on the tomographic system's essential functions, characteristics, constraints, and quality factors identified by the IAEA and related studies, the physical properties such as radiation resistance, waterproofing, heat dissipation should be first considered for mitigating the vulnerabilities of the designed tomographic system. Second, data transmitting and positioning methods should be additionally considered since the spent fuel assembly has to be moved and placed at the center of the tomographic system designed to rotate around an axis in the horizontal plane. Finally, safety concerns such as damage, degradation, failure, or derailment of the connected wire ropes also have to be considered for the system stability.

The purpose of this paper is to carry on the functional analysis of the tomographic system and to provide technical requirements for early-stage design. The further step is 1) to design and optimize the prototype tomographic system, 2) to perform the lab-scale feasibility test, and 3) to enhance imaging capability to detect partial defects on spent fuel items by applying an optimized image reconstruction algorithm.

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Keywords: Safeguards, Tomographic System, Spent Nuclear Fuel, Technical Requirements

Electrochemical Modeling of Partitioning of TRU From RE Elements

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In the pyroprocess of the spent fuel with the molten salt, partitioning of RE elements from the TRU elements is one of the important objectives when the recovered TRU is recycled as a fresh fuel for the transmutation system such as a fast reactor, because the amount of RE should be limited below 5 wt% in the fuel for the integrity and the performance of the fuel. For this purpose, there are processes to recover TRU elements from molten salt with minimal RE recovery is utilized such as the U/TRU recovery (so called electro-winning process), TRU drawdown and/or RE drawdown, and so on.

In the processes for the TRU recovery, liquid metal is considered to be used for the nonproliferation purpose, because plutonium is not separated from minor actinides such as neptunium, americium and curium when liquid metal is used as an electrode. The separation of TRU and RE elements in the molten salt and liquid metal can be predicted from the thermodynamic data, such as free energy of formation of chlorides, and activity coefficient of each element in the molten salt and liquid metal. The observed distribution of each element between two contacting substances was measured in the experiments in terms of distribution coefficients and separation factors for each element.

However, during the process, the amount of elements in the molten salt and the liquid metal changes continuously and it means that the thermodynamic prediction is not sufficient for the prediction of behavior of each element and instead kinetics should be considered together although modeling and simulation of multi-elements electrochemical reaction is not well developed yet.

In this study, electrochemical modeling of several case studies for the TRU and RE partitioning process were conducted to check the applicability of electrochemical models in those processes. The modeling and simulation can be complemented by accumulating real experimental data and lastly the semi-empirical model with the help of artificial intelligence can be a tool for the mass tracking of elements during the electrochemical process.

Keywords: Pyroprocess, Electrochemical, Modeling, TRU, Rare Earth

Status and Prospective of New Regulatory Framework for Spent Fuel Storage Cask

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It had been studied that the preliminary study to introduce the design approval system of spent fuel storage cask in the nuclear safety act since 2014. As a result, new design approval and fabrication inspection system are officially introduced in the nuclear safety act at the end of 2020. It will be enforced 6 months later. Until then its subordinate acts shall be amended including the nuclear safety act enforcement decree, enforcement regulation and regulations on technical standards for radiation safety control.

The aim of this study is to develop the draft of provisions related to the spent fuel storage cask the nuclear safety ; act enforcement decree, enforcement regulation, and the technical criteria for the design approval in the regulations on technical standards for radiation safety control.

In the nuclear safety act enforcement decree, it will be provided provisions of procedure and design criteria for obtaining the design approval, and provisions of application timing and acceptance criteria for fabrication inspection.

In the nuclear safety act enforcement regulation, it will be provided 13 items which shall be written in safety analysis report, and procedures for application of fabrication inspection.

In order to get a design approval, applicant's design shall meet the technical criteria that the 10 design criteria, 6 material criteria, 4 structure and performance criteria. The detailed design shall be described in safety analysis report compliance with the Nuclear Safety and Security Commission (NSSC) notice of standard format and content of safety analysis report for spent fuel storage cask.

Keywords: Fuel storage cask, Design approval, Fabrication inspection, Nuclear safety act

3분과

고준위폐기물처분 (Oral)

Long Term Geologic Evolution and Its Implication to the Safety Functions of a Potential High-Level Radioactive Waste Repository

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To assure the long-term post closure radiological safety of a potential high-level radioactive waste repository technologically and socially, it is important to develop the concept of the safety assessment from the beginning to get the feedback from stakeholders. The reference approach for the safety assessment assumes that the present-day environment shall prevail during the time of the interest for the assessment. But the natural environment has been evolving throughout the time. Therefore, it is also important to study the impact of the evolution of a repository system surrounded by geologic media. To assess the impact of the geologic evolution, the time frame for the assessment is a critical issue. The time frame for the assessment differs from countries. In this study, the authors propose that the future safety assessment shall cover the evolution of geologic media and engineered barriers for more than a million year. To predict the evolution for future one million years, the records of the revolution throughout the last 2.6 million years, the time span of the Quaternary Period are carefully reviewed. Generally, the Quaternary Period is divided into two periods; Holocene and Pleistocene.

The main geologic features during the Quaternary Period are summarized as (1) Cycles of glaciations and de-glaciations especially during the Holocene, (2) the sea-level change by the glaciations and de-glaciations, (3) the evolution of rift valleys with subductions and associated volcanic activities, (4) the long-term weathering and erosion, (5) active faults, and others.

The big issue is the glaciations during the Wurm Ice Age, the Last Glaciation Period. Literature surveys are conducted to understand the nature of the glaciation. The existence of Chukaryong Rift Valley between Seoul and Wonsan illustrates the impact of the geologic events. The actions of normal faults created the line shape rift valley during the Pleistocene. Now the activities of those faults are halted. Also, the area is covered with basalt throughout the magma eruptions. Many distinct geologic features are erased by weathering in many locations. Still, the line shape main structure is well preserved. The events of rift valleys shall be scrutinized to understand its impact to the long-term safety of a repository. Weathering and erosion is also important issues to fully understand that the 500 meter depth concept is solid enough to assure the proper pathway length of a radionuclide. In addition, available date for the active faults and their impact against the safety shall be reviewed.

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Keywords: Long Term Evolution, Total System Performance Assessment, Rift Valley, Weathering and Errosion

Estimation of Evaluation Parameters in Site Assessment for HLW Disposal in Terms of Rock Mechanics

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Site assessment procedure usually takes long period of investigation and is conducted in stages. At the same time, the procedure should take multidisciplinary evaluations into account so that it requires careful and systematic approaches. KIGAM had proposed a stepwise site characterization process, which is of multidisciplinary evaluation parameters in line with the feature of the assessment. In this paper, estimation method on several parameters, specifically intact rock properties in view of rock mechanical aspect, is introduced with some part of estimations made so far.

A comprehensive database on various intact rock properties, which can be utilized in stability and behavior analyses for HLW disposal repository, was constructed based on preexisting experimental results and references. Compiled database was statistically analyzed considering several factors, such as rock types, and tectonic provinces, then its characteristics and distributions were deduced further.

Meanwhile, two deep boreholes, 750 m of depth, were drilled in specific regions, where their bedrocks were found mainly as granite. Various investigations including in-situ and laboratory tests in multiple research fields were performed so as to characterized the regions and build a database on deep rock mass. Besides additional test results from the cored intact rock specimens, some part of the database, from which originated that specific region, were excerpted so that it was capable of conducting site-specific analyses. The excerpted database enables further site-specific characterizations, for instance Bayesian inference, as it could be used as prior information or distribution.

This paper is part of a research regarding site assessment for HLW geological disposal. It aims to make comprehensive database that can be used in the assessment and to characterize deep rock mass at multiple regions considering domestic geological features. As the research proceeds, results will have been updated consistently and presented accordingly.

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Keywords: HLW disposal, Site assessment, Evaluation parameters, Rock mechanics, Intact rock properties

Short-term Evolution of Transmissivity in Fractured Crystalline Rock Aquifer

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A blasting impact or stress redistribution after an excavation can have a significant effect on the original properties of a rock mass. The resulting change in the mechanical, hydraulic, thermal, and chemical behaviors of a rock mass around an excavation can significantly influence the overall performance of an underground facility. Therefore, it is vital to investigate the characteristics of an excavation damaged or disturbed zone for the safety and stability of a high-level radioactive waste (HLW) repository. In this context, this study analyzes the effect of the damaged zone near a borehole on the hydraulic characteristics of a crystalline rock aquifer which is preferred as a host rock for HLW repositories. For this purpose, pulse, slug, and constant head withdrawal tests were performed with various initial head displacements at 1.45 m intervals using a double packer system before and after the expansion of the BDZ-2 borehole located at the KAERI underground research tunnel (KURT). Hydraulic head response data from 171 hydraulic tests were analyzed by analytical solutions such as Cooper–Bredehoeft–Papadopulous, Jacob–Lohman, and straight line models. Reynolds numbers representative of the hydraulic tests were calculated and the influence of nonlinear flow on the hydraulic properties was evaluated. The analysis results showed that the estimated transmissivity can be distorted for high initial head displacements but the criterion can vary depending on the media. When the distortion was corrected, the transmissivities after the expansion were estimated to be up to two orders of magnitude higher than those before the expansion.

Keywords: Damaged zone, KURT, Hydraulic test, Nonlinear flow, Transmissivity, Crystalline rock

A Discrete Fracture Network (DFN) Model With Conditioning

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Conditional stochastic DFN modeling approaches such as bootstrapping, geocellular and hybrid conditional approaches were evaluated by comparing their simulation results on the geometrical and hydraulic properties of a network to those of the existing stochastic DFN model. The simulated spatially heterogeneous inflow rates into a disposal hole were also compared to each other. Relating to simulation of the network properties, the bootstrapping conditional method using a trace map was able to predict the density of percolating cluster closer to the real one than the other approaches, which seems that the geometric connection between boundaries is largely affected by the fractures near the boundaries and the bootstrapping method fix the fractures at the boundaries. The predicted results of equivalent permeability is considered to be that the conditional approaches cannot improve the existing DFN model, and this is because the suggested conditional approaches cannot reproduce the preferential flow path of the real fracture system. The inflow rate to a disposal hole was simulated with the conditional approaches and the existing one. The results show that the hybrid conditional stochastic DFN approach reduced variability of the prediction and could reproduce the tendency of the heterogeneous distribution of the hydraulic characteristics better than other approaches although the conditional approaches did not significantly improve the performance of the existing one.

Keywords: DFN model, Stochastic, Conditioning, Performance

New Analytical Solution of Multi-species Radionuclide Transport in Geological Barrier With Arbitrary Heterogeneous Aquifer

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This paper presents a semi-analytical procedure for solving coupled the multispecies reactive solute transport equations, with a sequential first-order reaction network in a geological barrier with arbitrary heterogeneous media using General Integral Transformation Technique (GITT). This proposed approach was developed to describe behavior of reactive multispecies transport on spatially or temporally varying flow velocities and dispersion coefficients with distinct retardation factors, which might be function of space and time. This proposed approach deals with general initial conditions, and arbitrary temporal variable inlet concentration as well as arbitrary heterogeneous media. The proposed approach sequentially calculates the concentration distributions of each species by employing only the generalized integral transform technique (GITT). Because the proposed solutions for each species' concentration distributions have separable forms in space and time, the solution for subsequent species (daughter species) can be obtained using only the GITT without the decomposition by change-of-variables method imposing the limitation of identical retardation values for all the reactive species by directly substituting solutions for the preceding species (parent species) into the transport equation of subsequent species (daughter species). The proposed solutions were compared with previously published analytical solutions or numerical solutions of the numerical code of the Two-Dimensional Subsurface Flow, Fate and Transport of Microbes and Chemicals (2DFATMIC) in all verification examples. In these examples, the proposed solutions were well matched with previous analytical solutions and the numerical solutions obtained by 2DFATMIC model. A hypothetical single-well push–pull test example and a scale-dependent dispersion example were designed to demonstrate the practical application of the proposed solution to a real field problem.

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Keywords: Analytical solution, Multispecies transport, GITT, 2DFATMIC

Rock Stress Estimation of Natural Barrier in Underground Disposal Research Facility: Establishing a Rock Mechanical Site-descriptive Model

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This study relates to an estimation of changes in stress field according to the construction of a geological disposal research facility at Korea Atomic Energy Research Institute (KAERI underground research tunnel, KURT). The 3D integrated geological model which can be represented by rock mechanical site descriptive model had been developed by containing rock mass characteristics. The mash domain was constructed based on the structural features including a number of major fault intersecting the site or area. The integrated information included topography, rock lithology and structural geology. Especially, the properties of the rock mass groups with fault zones were incorporated by assigning existing information from field data and drill core investigation. Based on well-structured mesh domain, the rock stress field was estimated using finite element method (FEM, COMSOL Multiphysics) and discrete element method (DEM, 3DEC). The stress distribution is pertained to the depth and the magnitude of the horizontal principal stress. As a first results, through comparison with actual stress regime from logged borehole data with hydraulic fracturing tests in underground research tunnel, the aspect of changes in stress field could be simulated and inversely estimated by adjusting applied far-field stress in the model. The majority of the analysis were achieved to understand the sensitivity with regard to the input parameters. And as a secondary result, using the representative regional model, the effect of construction and expanded size of the geological disposal research facility. The difference according to the analysis method had been also considered. The stress disturbance had been exhibited close to the discontinuities. As a consequence of this, the prevailing stress condition was different before and after the construction of the disposal research facility. The constructed rock mechanical model in this study will be utilized to investigate a hydraulic interaction between rock mass and ground water and characterize long-term evolution of natural barrier performance in disposal system.

Keywords: KURT, Natural barrier, Integrated geological model, FEM, DEM, Stress field

Machine-Learning-based Prediction Model of the Damage in KURT Granite Considering Various Acoustic Emission Parameters in Laboratory Scale

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Early prediction of near-field rock damage induced by cracks is an essential step in a deep geological environment such as radioactive waste disposal repository. Among monitoring techniques, acoustic emission (AE) is a powerful method of detecting crack growth of brittle material, especially in the pre-failure stage. To improve the reliability of rock damage prediction, considering various characteristics of AE signal is important, but this work is a difficult problem due to the complicated relationship between several AE parameters and the damage of rock. In this study, machine-learning (ML)-based methods are employed to deal with the complex features between several inputs and output. To achieving the purpose, uniaxial compression tests for KURT granite specimens are conducted with acoustic emission tests, simultaneously. Based on the testing results, an ensemble-based model (EBM) for the prediction of rock damage is developed and compared with a single-based model (SBM). On a laboratory scale, the results reveal that the generalization performance for the EBM is higher than the SBM. To capture model interpretability for the EBM, an analysis of parameter importance is carried out. It was confirmed that the cumulative AE absolute energy and initiation frequency give a positive contribution to the degree of rock damage. Although this study cannot be used directly to make predictions in situ, it provides useful information that the EBM is an applicable algorithm and what parameters are dominant for predicting the degree of damage. As further work, we will extend the study to engineering scale taking into account attenuation characteristics of rocks for practical purposes.

Keywords: KURT granite, Rock damage, Machine-learning, Acoustic emission parameter, Parameter importance

Review of Instant Release Fractions of CANDU and PWR Spent Nuclear Fuels for a Safety Case

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KAERI has developed a KBS-3 type geological repository for spent nuclear fuels with the KURT geological data. The results of radiological safety assessment after the closure of the repository strongly depend on the source terms. The source terms from spent nuclear fuels are divided into an instant release and a congruent release. According to the previous calculations, the instant release fraction (IRF) showed more significant effects than the congruent release even though the IRF is less.

According to Gray, it was common practice of performance assessment to assume that an average values 2% of radionuclides (e.g., ^{135}Cs, ^{129}I, and ^{99}Tc) were located in the gap and grain boundary regions. So far, the safety assessment for the geological repository carried out by KAERI assumed the IRF values based on the foreign cases. When the Safety Case was prepared by the implementing organizations, the IRF value of each radionuclide began to be assigned from the experimental results. The experimental results showed that IRF strongly depended on the linear power rating (LPR) and the discharge burnup of spent nuclear fuels. In this paper, the authors reviewed three Safety Case reports published in Sweden, Finland, and Canada. We summarized and analyzed the characteristics of the IRFs of major radionuclides including the relevant experimental results. The IRFs of the mobile nuclides such as ^{129}I and ^{135}Cs could be estimated from the measured Fission Gas Release (FGR).

Various kinds of spent fuels are being generated from CANDU and PWR nuclear power plants in Korea, and the burnups of PWR spent fuels continue to increase to improve the economics. It is known that a rim structure (High Burnup Structure, HBS) develops at the burnup exceeding 40 GWd/tU. The experimental leaching data with a high burnup spent fuel are few, and the radionuclides contained in the rim structure may contribute to the IRF is still not clear. It is strongly recommended that the leaching experimental results should be collected for the high burnup spent fuels to understand comprehensively the characteristics of IRFs of major long-lived radionuclides for a Safety Case in Korea.

Acknowledgements

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Keywords: Instant release fraction, Geological disposal, Spent nuclear fuel, High burnup

Copper Corrosion Modeling Based on Oxygen Diffusion in Aerobic Condition

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In this work, the corrosion behavior of copper disposal container was simulated in an aerobic environment. The long-term corrosion study of disposal containers is necessary for the design of container barrier to prevent the leakage of nuclear species throughout the guaranteed period, and for the safety assessment of disposal sites. The approach to corrosion modeling was based on the assumption that corrosion occurs only when dissolved oxygen passes through the buffer protecting the disposal container and arrives at the copper surface. Therefore, the diffusion coefficient of dissolved oxygen in buffer materials is key parameter. And the diffusion coefficient of copper cation in the passive oxide film was also considered. Therefore, it was interpreted that dissolved oxygen and copper ions meet in the porous oxide layer, and which result in a new corrosion product on the copper canister.

Corrosion modeling was performed by Visual basic together with Microsoft Excel. It was found that the copper cation diffusion in oxide film was a limiting factor in the initial stage, but the diffusion of dissolved oxygen through the buffer was a limiting factor after sufficient swelling of bentonite buffer as result of the modeling. Current modeling results showed a high corrosion rate up to about 200 hours, but since then, a constant corrosion rate was confirmed. However, since temperature has a large effect on the actual corrosion, we are attempting to apply the temperature change around disposal canister caused by decay heat in the modeling.

Keywords: Copper, Corrosion, Modeling, Aerobic, Diffusion, Simulation

Time-Resolved Luminescence Properties of U(VI) Species Adsorbed on Ca- and Na-Montmorillonites Produced From Domestic Bentonite

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Migration of uranium in groundwater is influenced by sorption to several clay minerals and metal oxides. Although the sorption properties of U(VI) onto a domestic bentonite were investigated in the previous study, the microscopic sorption characteristics at a molecular scale have not been fully identified yet. In the present study, time-resolved laser fluorescence spectroscopy (TRLFS) was performed for the adsorbed U(VI) on Ca- and Na-montmorillonites. The luminescence properties (luminescence peak position and lifetime) of these samples are compared with those of silica and alumina.

Ca- and Na-montmorillonites: 2:1 structured alumino-silicate clay minerals saturated with calcium and sodium, respectively, were produced from KJ-II Gyeongju bentonite by using separation, refinement, washing, and saturation processes. Commercially available silica and alumina powders were used for the uranium contained oxide suspensions. The solid concentrations of the montmorillonites and oxide suspensions were approximately 1–3 g/L. Uranium adsorption reactions onto these samples were performed in 30 mL of mixed solution (3–10 M UO_2^{2+} and 0.1 M $NaClO_4$) with a pH of 7.5. Pulsed Nd-YAG laser at 266 nm (Continuum Minilite) was used as the light source of TRLFS. The laser pulse energy of 0.1 mJ was fixed for all measurements to avoid the photochemical reaction. The luminescence spectrum was recorded using a gated intensified charge-coupled device (Andor, DH-720/18U03 iStar 720D) attached to the spectrograph (Andor, SR-303i).

It is observed that the luminescence characteristics of Ca-montmorillonite are similar to those of silica samples. In contrast, the characteristics of Na-montmorillonite are similar to those of alumina samples. Based on the previous extended X-ray absorption fine structure spectroscopy and TRLFS results, possible U(VI)-silica, U(VI)-alumina and U(VI)-montmorillonite surface interactions are discussed.

Keywords: Sorption, Montmorillonite, Silica, Alumina, TRLFS

Preliminary Evaluation of Mineralogical Alteration and Sorption Property of Bentonite

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To evaluate the mineralogical alteration and sorption properties of bentonite under the condition of high temperature and pressure that reflects the deep geological condition of high temperature (100℃) and pressure (50 bar) those can reflect deep geological disposal condition, we conducted preliminary short term (100 days) reaction tests using powdered Ca-bentonite and synthetic KURT groundwater in the hermetically-closed batch reactors. Through the tests, the synthetic KURT groundwater was controlled to contain 0.7, 50, 50 mg L^{-1} in the concentration of nuclides (U, Cs, and I) respectively, and the effects of particle size (10-75 m) and montmorillonite grade (60-85%) of the bentonite during the reactions were investigated. Due to the bentonite alteration (mineral degradation), the major constituents in the aqueous phase showed a slightly increasing trend, but the trends under different experimental conditions were not clear to be interpreted by relatively short reaction time (100 days). The XRD results of the bentonite after the reaction showed that the basal refraction intensity and the peak of montmorillonite were decreased and shifted from 1.44 to 2.00 Å, respectively, which indicates that the montmorillonite in bentonite was saturated by water and/or partially collapsed during the reaction. And the nuclides (U, Cs, and I) in the groundwater were mostly adsorbed and removed throughout the reactions. The experimental results on mineralogical alteration and sorption capacity of bentonite may helpful to be used for a reactive transport scenario in which the nuclides released from the waste canister migrate in the bentonite buffer. For further study, yearly-based long-term experiments to prove the longevity of bentonite have been designed.

Keywords: Nuclide ion, Bentonite, Sorption, Mineralogical alteration

Leaching Behaviors of Cs (Ⅰ), Sr (Ⅱ), Co (Ⅱ), and Eu (Ⅲ) From Solidified Portland Cement Matrix

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A numerous low-level and intermediate-level radioactive wastes might be generated from the decommissioning processes of nuclear facilities need to be immobilized to prevent the release of radionuclides from the wastes under the disposal condition. In this study, the Portland cement was used as an immobilization agent for the nuclides such as Cs, Sr, Co, and Eu, and their leaching behaviors were investigated. Throughout the study, the leaching experiments were performed according to the ANS 16.1 “Measurement of the leachability of solidified low-level radioactive wastes by a short term test procedure”. The Cs, Sr, Co, and Eu were in order of the leaching rates which decreased with an increasing of leaching time. The 41.4% of Cs was especially leached out for 90 days although the others were little leached less than 1.5%. In spite of high leaching rate of Cs, the Portland cement can be considered as an effective matrix potentially for immobilization of Cs, Sr, Co, and Eu, since the leachability indices in all cases exceeded the threshold value of 6 as waste acceptance criteria. The results of present study can be used in the safety assessment of a LILW repository.

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Keywords: LILW, Portland cement, Leachability, Safety assessment, ANS 16.1

Effect of Temperature History on Integrity of HLW Disposal Repository

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High-Level radioactive waste (HLW) is recommended to be disposed in deep geological condition for separating nuclides from biosphere over tens of thousand years. Thus, it is essential to estimate long-term integrity of the repository with consideration of disposal conditions. The disposal repository is composed of HLW, canister which contains HLW, buffer which postpones the groundwater from reaching canister and release the heat from the HLW, natural barrier which indicates near-field rock mass, backfill, plug and so on. Even though HLW is cooled over several decades in pool, it still emits high enough heat. Emitting heat rises the temperature of buffer and near-field rock mass. And near-field rock mass is saturated because the repository is located in a few hundred meters below the water table.

The influence of the temperature and water content on natural barrier is hard to be solely considered because the characteristics of the natural barrier have inherent fractures and fissures. In this study, concrete specimens are cured during 28 days to demonstrate uniform intact rock specimens and the temperature is controlled after curing as 15℃, 45℃, and 75℃ which represents the saturated rock mass without HLW, normal condition after emplacing HLW, and accident situation which exceeds the designed temperature, respectively.

The specimens exposed in 15℃ (SC15) have about 10-20% higher water contents than the specimens exposed in 45℃ (SC45), and 75℃ (SC75). Although it is known that the water contents and uniaxial compressive strength have a reciprocal relationship, SC15 give about 10% higher uniaxial compressive strength than SC45 and SC75. This indicates that the integrity of brittle material in repository can be significantly degraded when it is exposed in high temperature.

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Keywords: High-Level radioactive waste, Disposal repository, Uniaxial compressive strength, Temperature, Water content

Numerical Modeling of FE Experiment at Mont Terri Underground Rock Laboratory in the DECOVALEX-2023

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It is essential to understand the complex thermo-hydro-mechanical-chemical (THMC) coupled behavior in the high-level radioactive waste disposal system to secure the repository's long-term safety. The heat from the canister induces thermal pressurization and vaporization of groundwater in the geological media, and the inflow of groundwater changes the saturation of the engineered barrier system. The saturation variation affects the heat transfer and multi-phase flow in the buffer material. Due to the complexity of the coupled behavior, a numerical simulation is a valuable tool to predict and evaluate the THM interaction effect on the disposal system and safety assessment.

To enhance the knowledge of THMC coupled interaction and modeling techniques in geological systems, DECOVALEX, an international cooperative project, initiates from 1992. The current phase, DECOVALEX-2023, consists of 7 tasks, and among them, Task C is to model the Full-scale Emplacement (FE) experiment at the Mont-Terri underground rock laboratory. FE experiment was based on Nagra's reference repository design at a 1:1 scale. The experiment aimed to investigate the THM coupled effects on the surrounding rock and evaluate emplacement procedures for underground conditions. Based on the experimental data, nine groups (BGR, BGE, CAS, ENSI, GRS, KAERI, LBNL, NWMO, Sandia) modeled the THM processes using the numerical codes to understand pore pressure variation and two-phase flow systems in the buffer material.

We used OpenGeoSys for the numerical modeling in the task, and Richard's flow model was applied for the multi-phase flow simulation. For the preliminary study of the task, a simple two-dimensional model was prepared with a domain size of 50 by 50 meters. A single heater was emplaced horizontally on the center of the domain, and the bentonite block and granular bentonite filled the engineered barrier system. Based on the numerical model, we performed a series of simple sensitivity analyses to investigate the affecting factors on the flow systems for various hydraulic parameters. Pore compressibility and porosity were revealed as the dominant factors affecting the pore pressure variation in buffer and geological systems.

Additionally, the flow process induced by the capillary effect was dominant in the near-field of the canister, while thermal pressurization of fluid was substantial in the far-field area. The developed model will be applied to a full-scale three-dimensional numerical simulation, and the multi-phase flow model considering capillarity as a primary variable will also be compared for efficient computing systems.

Acknowledgements

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Keywords: DECOVALEX-2023, FE experiment, Mont-Terri URL, OpenGeoSys

3분과

고준위폐기물처분 (Poster)

Limitation and Relevance of Natural Analogue Studies for Radionuclide Behaviors in HLW Disposal

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In this study, a short overview of limitation and difference, and relevance – lesson learnt of natural analogue studies is presented focusing on radionuclide behavior of high-level radioactive waste (HLW) repository. One of the most challenging aspects in a deep geological repository for HLW is to apply experimental data obtained during short-time scales to long-time scales required for the safety assessment. Natural and anthropogenic analogues can provide important information for radioactive waste forms and performance of radioactive waste repositories over the geological time scale. However, using the results from analogue studies in the safety assessment models should be careful due to some limitations and differences. It is impossible to define initial conditions of natural and anthropogenic analogues and their materials and environments to be studied will be different from those considered in real repositories.

Although analogues provide valuable data to the safety case as one of additional evidences, we can not depend on the analogues as the only evidence because it is very difficult to find analogues similar to the processes to be occurred in repositories. In order to utilize the data from natural analogue studies, therefore, the limitations and the differences of natural analogue studies with real disposal systems should be investigated and then used by analyzing the relevance and applicability of the natural analogue studies. Implementers and researchers should be careful in identifying overestimated results from natural analogue studies for assuring the public reliability for the safety and performance of HLW repositories. Therefore, this study will contribute to investigate the limitations and applicability of existing results of natural analogue studies necessary for the development of a safety case for HLW disposal system.

Keywords: High-level radioactive waste, Safety case, Natural analogue study, Uranium deposit, Radionuclide migration and retardation

Optimization of Spent Fuel Assemblies per Canister Based on Decay Heat for the Optimum Design of a Deep Geological Repository

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Various kinds of spent nuclear fuels (SNF) are being stored in the spent fuel storage pools. They will be disposed into a deep geological repository (DGR) at the end. According to the conceptual design of a DGR for SNFs, four spent fuel assemblies (SFA) are supposed to be emplaced into a canister. From this concept, the disposal area is estimated considering spacing between deposition holes and disposal tunnels that are determined by thermal analysis using the decay heat of a reference SFA. If we use the actual decay heat for each SFA instead of using a reference SFA during the design process, however, we can reduce the disposal area.

From this point of view, we developed an optimization algorithm for the combination of SFAs per canister to reduce the overall decay heat from the canisters using the actual decay heat data of each SFA. The decay heat of each SFA is estimated using regression equations considering the characteristics data of each SFA such as burnup, discharge time, cooling time, and disposal schedule. During the combination of SFAs per canister, it is assumed that four SFAs in a canister are discharged from the same reactor. Based on the number of total SFAs to be disposed, the number of SFAs per canister, and the number of canisters to be disposed per day, overall disposal schedule is set up with blank canisters. Initially, then, all SFAs are randomly assigned to canisters. During the optimization process, the SFA having the maximum decay heat in the canister having the maximum decay heat and the SFA having the minimum decay heat in the canister having the minimum decay heat are exchanged, and the decay heats for both canisters are recalculated considering the changed disposal schedule. This optimization process is iterated until a user-defined criteria is satisfied.

We developed a computer program using MATLAB based on the optimization algorithm. The stability of the program was confirmed by analyzing iteratively the overall distribution of decay heats from the canisters. Then we applied the program to the disposal scenario suggested in the research program for the design improvement of a DGR for SNFs. From the results, it could be confirmed that the overall decay heats from the canisters became evenly distributed and the maximum value was quite smaller than that of reference SFA by the optimization.

Keywords: Spent Nuclear Fuels, Deep Geological Repository, Decay Heat, Optimization

Evaluation of Thermal Properties for the Gap-Filling Material

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The high-level radioactive waste (HLW) produced from nuclear power plants is disposed in a rock-mass at a depth of hundreds meters below the ground level. Since HLW is very dangerous to human being, it must be disposed of safely by the engineered barrier system (EBS). The EBS consists of a disposal canister, backfill material, buffer material, and so on. When the components of EBS are installed, gaps inevitably exist not only between the rock-mass and buffer material but also between the canister and buffer material. The gap can reduce water-retarding capacity and heat release efficiency of the buffer material, so it is necessary to fill the gap with some adequate materials. Even though there has not been enough researches on the investigation of gap-filling materials, especially in Korea.

In this reason, this research analyzed the gap spacing effect on the peak temperature of the bentonite buffer considering KRS (KAERI Reference disposal System) based on numerical analysis of heat transfer module. The gap between the canister and buffer material had a minor effect on the peak temperature of the bentonite buffer material, but there was 40% difference of the peak temperature of the bentonite buffer material because of the gap existence between the buffer material and rock mass. Furthermore, this paper evaluated thermal conductivity and capacity of the candidate gap-filling material based on the line source theory. The clay granular was used, and particle size was around 1~3 mm. The clay granular was filled in the steel cell, and bulk density was around 1,000 kg/m^3. The thermal conductivity value was 0.158 W/(m·K), and volumetric heat capacity value was 1.274 MJ/(m^3·K). It is essential to investigate various candidate materials for the gap-filling.

Keywords: Gap-filling material, Clay granular, Thermal properties

Lessons Learned From Posiva's Safety Case Report on Canister Materials Stability

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Finland is far ahead in the field of radioactive waste management. Finnish regulator approved Posiva's waste repository plan and issued a construction license for the repository for the spent nuclear fuel (SNF) disposal. Posiva Oy's Safety Case report on the disposal of spent nuclear fuel at Olkiluoto was essential to make this happen. In Posiva Oy's Safety Case report portfolio, Complementary Considerations (Posiva 2012-11) was one of the prerequisites. The objective of Complementary Considerations is to enhance confidence in the outcomes of the safety assessment for the disposal repository to be constructed. The main emphasis is on the evidence and understanding that can be gained from observations from natural and anthropogenic analogues relevant to the repository including SNF, EBS, NBS, radionuclide migrations, etc.

In this study, we investigate how the Posiva Oy's Safety Case – Complementary Consideration report is organized and focused on the evidences for the suitability of the repository materials. In natural analogue study on the canister materials stability, the canister corrosion processes considered include general corrosion under oxic and anoxic conditions, localized corrosion, microbially influenced corrosion, and stress corrosion cracking. For the representative canister materials (*i.e.*, copper and iron), the natural analogue study on the copper archaeological analogues, copper geological analogues, and iron analogues were well described. This can be used as a role model for Korean natural analogue study on the canister materials.

Keywords: Natural Analogue, Copper Analogue, Iron Analogue, Safety Case, Canister Material Stability

First Step of Buffer Durability Study in Natural Analogue for Waste Disposal

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Currently, countries planning direct disposal are working with the design of disposal systems. Finland, in particular, has secured the site of the world's first spent fuel research disposal site, and is known to have similar conditions to geological conditions based on crystalline rock in Korea. POSIVA, Finland's radioactive waste management agency, is planning ONKALO, a deep geological disposal research demonstration facility, while also focusing on improving reliability of disposal systems. For one example, POSIVA has being actively utilized natural analogue study in reports describing supplementary considerations for safety case development. This is because natural similar studies are considered as one of the integrated parts of disposal programs in most countries considering the disposal of radioactive waste.

Natural Analogue study, according to the IAEA, is defined as similar approaches to investigating the natural occurrence of substances, conditions, and processes equal to or similar to those known or predicted to occur in certain parts of the disposal system. In addition, natural analogue research is an area where not only geological research but also complex academic cooperation such as materials and history are required. Therefore, it has been consistently conducted through global cooperation for more than 30 years, but domestic systematic research has not been attempted in Korea yet. It means more effort is needed, such as developing a methodology for how acquired natural analogue information can be applied into improvement of reliability for safety case.

In this study I would like to conduct a natural analogue study of buffer material, an important far-field element being able to protect humans from the danger of high-level radioactive waste, in disposal system. As a first step for the study, the approach of natural analogue for buffer material in the safety assessment at Olkiluoto should be analyzed. And based on this, I hope that the direction of domestic natural analogue research will be established.

Keywords: Natural analogue, POSIVA, Disposal system, Safety assessment, Buffer material

Development of Numerical Model for Sulfide-induced Corrosion of Copper via MOOSE

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Corrosion of copper canisters can reduce life-time of canisters in deep geological repository significantly, hence corrosion is the major issue for construction of deep geological repository for the whole world. Sulfide is major oxidant spices to cause corrosion of copper with respect to long-term corrosion in deep geological repository. Despite of importance of sulfide, there is no transient numerical model to predict sulfide-induced corrosion of copper in South Korea. Therefore, authors present numerical model to predict sulfide-induced corrosion of copper in anaerobic condition. A transient numerical model is developing based on mixed-potential theory and finite element method via MOOSE (Multiphysics Object-Oriented Simulation Environment). This model implements Multiphysics which includes reactive transport, and thermal analysis. Experimentally reported corrosion potential and corrosion product film of copper could be reproduced through our numerical model. The developing numerical model will be applied to simulate corrosion of copper for deep geological repository in South Korea.

Keywords: Mixed-potential theory, FEM, Sulfide, Copper, Corrosion, Deep geological repository

Scenario Analysis for Thermo-Hydraulic Behavior in Deep Geological Repository

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Thermo-hydraulic behavior in deep geological repository is a main issue for temperature distribution and also affect to copper canisters corrosion behavior. Therefore, this study aimed to figure out the various thermo-hydraulic behavior within the deep geological repository. Herein, authors present new numerical model for thermal-hydraulics analysis for deep geological repository system based on finite element methods. The following two scenarios were designed to calculate temperature of an existing design of deep geological repository with respect to design criteria of maximum temperature of bentonite: Scenario1 assumed that the degree of saturation in the host-rock always saturated, and Scenario2 considered the underground water behavior within the host-rock. The existing design of deep geological repository system was not satisfied the design criteria of temperature (100^{o}C)) from the result of scenario 2 (95^{o}C). So, it can induce the safety issue related with temperature. Therefore, in this paper, proper repository design for the temperature problem is suggested.

Keywords: Deep geological repository, Thermo-hydraulic behavior, Hydraulic behavior scenario, Safety issue

Continuous Production of SiO_2-Al_2O_3-P_2O_5 by Using Aerosol Process System to Dechlorinate and Vitrify Radioactive Salt Waste From Pyroprocessing

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The recycling of spent nuclear fuel via pyroprocessing involves a series of electrolytic processes that utilize metal chlorides as electrolytes. The resulting salt waste is radioactive due to the presence of fission products, and should be treated carefully. Methods of treating radioactive waste include immobilization, one of which is immobilization via dechlorination. In previous study, SAP (SiO_2-Al_2O_3-P_2O_5) was developed that could be used as both dechlorination and vitrification reagent. The low compatibility between alumino-silicate glass and alumino-phosphate glass has limited SAP production to a conventional sol-gel process. However, this process is associated with a limited batch size and requires a minimum processing time of 5 days. The present study investigated the development of an aerosol process system for the continuous production of SAP, with the potential of scaling-up for mass production. The process is comprised of the atomization of precursor solution into droplets by using a droplet generator. The droplets are transferred directly through a heating furnace by a carrier gas, where the solvent evaporates and intra-particle reactions occur to form the product. This aerosol process has several advantages over conventional material processing techniques like the sol-gel process. Most importantly, the aerosol process allows the formation of a highly pure product powder. Liquid and solid-state process require an additional milling process to obtain the powder, thereby introducing some impurities. However, the purity of the aerosol-processed powder is only dependent on the purity of the precursor solution. Further, the particles produced by aerosol process are more uniform in size and composition compared to those produced by other processes. Multi-component materials can also be produced by aerosol process, as each droplet contains precursors with the same stoichiometry as the desired product. From the present study, the glass form of aerosol-processed SAP was successfully produced by using an aerosol process system, and no additives were required to promote vitrification or prevent crystallization. Through the aerosol process system, pure SAP was produced by using a precursor with Si:Al:P ratio of 1:1:2 ($2SiO_2$-Al_2O_3-$2P_2O_5$).

Keywords: Pyroprocessing, Radioactive waste, SAP (SiO_2-Al_2O_3-P_2O_5), Aerosol processing

Current Status of the IAEA URF Network and the Results of the 19th Annual Meeting

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A URF (Underground Research Facilities) or URL is an underground facility in which site characterization and testing activities are carried out along with technology development and demonstration activities in support of the development of deep geological repositories for radioactive waste. All URFs play an important role in the development of deep geological repository systems for the disposal of long lived and high level radioactive waste, both from a scientific and technological point of view and for building public confidence.

The IAEA Underground Research Facilities Network (URF Network) was formally initiated in 2001. The URF Network's goal is to encourage the development of safe, sustainable and effective geological disposal programmes around the world through demonstrations of technology, improved training and enhanced communications between participating organizations (TOR of URF Network, IAEA, 2016). The scope of the URF Network is to provide and maintain a community of practice and learning for geological disposal, platform to assess and share best practices in developing, evaluating and implementing geological disposal solutions, and platform which emphasises the role and use of URFs to support successful geological disposal implementation. At present, the URF Network members are from 34 member states.

The URF Network organizes an annual meeting, whose primary purpose is to tailor and plan the URF Network activities to meet timely URF Network members' needs. The 2021 Technical Meeting of the URF Network, the 19th in the series, on the theme of "Global Progress in Developing Geological Disposal Solutions", was hosted by the IAEA through a virtual platform, from 22 February to 4 March 2021. The purpose of the meeting was to provide an overview of recent developments and progress made in various national DGR programmes, an overview of key experiments conducted or planned in URFs, as well as to explore options for cooperation between URF Network members. There was a session to review recent Network activities and discuss the future strategic orientation of the programme of work by the Network in 2021 and beyond. There was also a session to discuss progress and explore shared interest among URF network members for future topical cooperation, such as on the URF Compendium, the DGR Roadmap, the overview of site selection criteria and the planning of DGR RD&D programmes.

Acknowledgements

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Keywords: URL (URF), Deep Geological Repository (DGR), URF Network

Case Study on Long-Term Evolution of the Surface Environment by Climate Change for Deep Geological Disposal of Radioactive Wastes

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For examining performance and safety of the deep geological disposal, long-term time scale should be considered because it takes hundreds of thousands of years for high-level radioactive wastes to have little radiological effect on surface ecology. During these long-term time scale, the surface environment as well as deep geological conditions could be changed. The changes of surface environment including hydrological aspects such as spatial distribution of lakes and rivers have influence on safety assessment by changing transport path of radionuclides from the radioactive wastes and performance assessment by changing hydrogeological and hydrochemical conditions interacting surface water bodies.

Changes in climate have been brought up as one of import factors causing the long-term evolution of the surface environment. Therefore, in some countries developing the deep geological disposal considered changes in past and future climate to show long-term safety of the geological repository.

In high latitude regions such as Sweden and Finland, changes in climate may be a key to glaciation. These two countries identified periodical variations of climate. Then they estimated spatial distribution of past ice sheets and expected future scenarios of developing glacier. These were used to expect changes of terrestrial area including coastline change by sea level variation, and changes of loading by ice sheet on surface and underground including repository depth.

In Japan, there are two underground research laboratories with different bed rocks and two separate studies for long-term evolution of past climate change were conducted using field data obtained from the two laboratories. There was little effect of glaciation in two study areas. After the last glacial period, the sea level has been lowered and terrestrial area has been expanded in the cold region. On the other hand, the changes of a precipitation and a recharge rate have transformed topographical relief in the temperate region.

Although climate change occurs on global scale, there are regional differences in the manifestation of the effects of climate change. Therefore, the characteristics of climate change around the Korean Peninsula should be examined, and the methods for analysis of the characteristics and approaches to reflecting them to estimate long-term evolution of the surface environment were required.

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

Preliminary Study of Cesium Behavior Within Rock Fragment Under Deep Geological Condition

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In order to understand the cesium behavior through a natural barrier in terms of the HLW geological disposal system, preliminary short-term (100 days) reaction tests were carried out using the granitic rock fragments (5Ｘ7 cm) and synthetic groundwater collected from the borehole in KURT (KAERI Underground Research Tunnel). Prior to the reaction, the rock fragments were prepared separately fresh (non-altered) and fractured (altered) ones and the KURT groundwater was synthesized by adding non-radioactive cesium compound adequately to contain 50 mg L^{-1} in concentration. All of the reactions were controlled under the condition of 100℃ and 50 bar in temperature and pressure, respectively, in a hermetically-closed batch reactor. After the reaction, above all, the mineralogical analysis result showed there was no significant alterations in the mineral composition both of rock fragments during the reaction indicating the insufficient reaction time for mineral degradation in a rock fragment. So, yearly-based long-term tests to investigate the rock weathering and its retardation capacity for nuclides under the geological disposal environments have been designed. Throughout the reaction a substantial amount of cesium was removed by sorption onto the rock fragments. Through there were no quite differences in a removal efficiency, the distribution coefficients of fractured rock fragment (840～930 mL/g) were calculated slightly higher than those of fresh rock fragments (760～810 mL/g). Actually, some of secondary minerals are distributed along the fractured zone within the rock fragment, which is well-known as a good natural sorbent compared to the primary minerals. These preliminary results for Cs sorption onto geo-media are coincided with other previous studies on high sorptive property.

Keywords: Rock fragment, KURT, Mineral alteration, Cesium, Sorption

Numerical Approach for COMSOL Multiphysics Simulation in APro

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KAERI recently developed a process-based total system performance assessment model for a geological disposal system, called APro. APro has a framework to simulate the radionuclide transport considering thermal, hydraulic, mechanical and geochemical changes in the near and far field of the geological disposal system. This could be implemented by linking PHREEQC specialized in geochemical reaction to COMSOL Multiphysics, which is limited to solute transport simulation considering thermal, hydraulic and mechanical changes, in MATLAB interface. And the SNIA (sequential non-iterative approach) of the OS (operator splitting) technique was applied to couple the simulations performed independently in COMSOL Multiphysics and PHREEQC.

APro aims to be used for the total system performance assessment of long-term evolution scenarios expected to occur in a geological disposal system. The actual repository will have more than 10,000 boreholes, and the problem of simulating the radionuclide transport from such multiple boreholes over a period of more than 100,000 years, considering coupled processes, can be computationally intractable. In particular, since solving the nonlinear PDEs (partial differential equations) in COMSOL Multiphysics accounts for the majority of APro's computational burden, optimizing the numerical approach in the COMSOL Multiphysics is essential. COMSOL Multiphysics constructs linear system equations in a form of matrix equation (Ax=b) from the nonlinear PDEs through the FEM (finite element method), Newton-Rahpson method and time-stepping method. Then solutions can be obtained either at once (fully coupled) or sequentially and iteratively by governing equations (segregated) via LU decomposition (direct solver) or gradient method (iterative solver).

This study compares and analyzes robustness and efficiency (computation time, memory usage, possibility of parallel computation) of the numerical approaches for the COMSOL Multiphysics simulation in the near field. It aims to select a numerical approach suitable for solving large scale problems and to apply it to the development of new high-performance numerical approaches.

Keywords: APro, Total System Performance Assessment, Geological Disposal System, COMSOL Multiphysics Simulation, Numerical Approach

Basic Research on Methods of Rock Simplification and Classification Applied to Disposal Study: a Case Study in Finland

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Determining the rock types and grasping its distribution is important for reconstructing the temperature, pressure, and paleostress change during the rock formation, and is essential for estimating the evolution of the natural barrier. However, the criteria for rock classification for the radioactive waste disposal repository have not been accurately established in domestic studies. So, we conducted research on the methodology of classifying rock types in the case of Finland prior to further analysis such as thin section studies, geochemical analyses, and age datings. This research also can contribute to develop the precise three-dimensional geological model using borehole survey of the KURT area. The rock types in Finland are classified based on the geological mapping of the outcrops, investigation trenches, and the logging of the drilling sites. The data from these methods are grouped in A (linear sampling), B (areal sampling) and C (volumetric sampling) according to the data and level 1 (general level), level 2 (systematic level) and level 3 (research level) according to the levels of investigation. Therefore, the data are classified into total 9 types from A1 to C3. Level 1 (A1, B1 and C1) includes almost all the data for level 2 (A2, B2 and C2) and level 3 (A3, B3 and C3), so investigation method for level 1 data is important. To construct qualitative data for level 1, detailed investigation of rock observation and classification are needed. This study is basic research on how to simplify the data (e.g., method of simplify the distribution of the rock gathered through geological investigation according to the properties of each rock), to improve reliability by reducing analytical errors (e.g., comparative analysis of resolution using color histograms of borehole images), and to establish criteria for classification of the rock by using data sources of level 1.

Keywords: Geological data, Rock classification, Natural barrier, Radioactive waste disposal

Analysis on the Current Status of Domestic and Foreign Radionuclide Sorption Databases for the Safety Assessment of Spent Nuclear Fuel Disposal

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Countries with nuclear power plants around the world are considering or constructing repository for safe disposal of radioactive waste. The most important purpose of radioactive waste disposal is to isolate nuclides in radioactive waste disposal site, and to minimize their migration to the ecosystem when nuclides are released. In particular, the migration and retardation of nuclides in the underground environment is greatly influenced by the interaction between the nuclides and the surrounding host rocks, groundwater, and fracture filling minerals, that is, sorption. Therefore, sorption is an important factor in evaluating the safety assessment of radioactive waste disposal in a deep underground disposal environment. Sorption of nuclides is greatly influenced by various geochemical factors such as mineralogy, pH, redox potential, carbonate concentration, ionic strength, dissolved organic matter, and concentration of nuclides. Since the sorption mechanism is very complex, it is very difficult to select a representative sorption value (K_d, distribution coefficient) under specific geochemical conditions or to obtain necessary sorption values from a vast amount of sorption data. For this reason, various types of sorption databases (SDB) have been developed over the past several decades. The purpose of this paper is to analyze the current status of domestic and foreign SDBs for safety assessment of spent nuclear fuel disposal and present the future plan for the development of domestic SDB.

Keywords: Radionuclide, Safety assessment, Sorption, Database

Improvement of Three-Dimensional Geologic Modeling of the KURT Site for Analyzing Long-Term Evolution of the Natural Barrier

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Three-dimensional geologic models of the KURT site were started to build more than a decade ago to characterize the geological features for high-level radioactive waste disposal research. To build three-dimensional geologic models, several geological investigations such as geophysical survey and borehole drilling were performed. Numerous raw data were collected and computerized to use as input data for geologic modeling. Three-dimensional geologic modeling was then performed. In this study, three-dimensional geologic modeling of the KURT site is performed again to improve previous geologic models for long-term evolution of the natural barrier research. Before utilizing previous input data and geologic models, improvements are performed in terms of the following three aspects in order to build three-dimensional geologic model more accurately and precisely for further studies. First, digital elevation model of the study area is improved. Although digital elevation model used at the previous geologic modeling has numerous data point, replacing digital elevation model with updated version has advantages in terms of precision and reliability. Second, coordinates of data such as locations of geophysical survey line and borehole are adjusted. Some input data shows a discrepancy because of the raw data is collected from various sources and computerized to use as input data. Third, modeling domain is optimized. The location of the domain boundary and size of modeling domain is determined by the objective of study. As the objective of research project is changed to long-term evolution of the natural barrier, suitable three-dimensional geologic modeling domain is changed. The three-dimensional geologic model built in this study can be utilized as a base model of the KURT site for development of an assessment technology for the performance of long-term evolution of the natural barrier research.

Keywords: Radioactive waste disposal, Natural barrier, Geologic modeling, Long-term evolution

TBM Excavation Method for High-level Radioactive Waste Repository

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The cumulative amount of high-level radioactive waste (HLW) generated by nuclear power plants stored on the ground has almost reached saturation. For the safe and stable disposal of high-level radioactive waste, an in-depth HLW repository is needed to store radioactive waste deep underground. The HLW repository is mainly built in a structurally safe rock layer that is 800 m or more deep underground. Machines to transport a canister which is a storage container containing radioactive waste put the canister in the hall are used, so the space is relatively large considering the working space. In addition, since radioactive waste must be blocked from groundwater to prevent the spread of radioactive contamination, and the temperature of the HLW repository must be maintained within an appropriate temperature range by dissipating decay heat to the surroundings, joints and discontinuities in the near-base rock of the repository adversely affect the repository. The construction of the crown of a HLW repository is also important because the backfill minimizes the voids at the interface between the repository and the near-base rock mass to remove the air layer so that heat can spread to the surroundings well. Thus, HWL repository should be constructed with minimal impact of excavation on the rock so that engineered barrier systems and natural rocks interact well with each other. For these reasons, the TBM method is more suitable for excavation of a HLW repository than a blasting method, which causes unintended cutting sites and various microcracks due to relatively large vibrations and explosions which affect the rock structure. Therefore, this study deals with a method of an appropriate TBM operation for the HLW repository in rocks based on several TBM excavation studies in a rock mass.

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Keywords: High-level radioactive waste repository, Rock, TBM, TBM excavation

Influence of Bentonite Alteration on Radionuclide Sorption

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In the underground disposal of high-level radioactive wastes, bentonite would be used as a buffer material around the metal canister. The bentonite was composed of montmorillonite and minor minerals such as quartz and feldspar. The bentonite will be gradually altered from its initial state over time. Particularly, it is affected by metallic components of iron oxide minerals in bentonite. In addition, indigenous bacteria that exist in bentonite will also play as a biocatalytic role to alter the bentonite by reducing the oxidized metallic components.

In our experiment, we examined the alteration of bentonite that was incubated for several months under anaerobic conditions. Hematite, one of the iron oxide minerals in bentonite, was gradually dissolved by a microbial reductive reaction during the incubation. The reduced Fe^{2+} from hematite was released and replaced the bentonite interlayer cations such as Ca^{2+}. In addition, as the structural Fe^{3+} of montmorillonite was reduced by the microbial reductive process, the layer charge of montmorillonite was changed, affecting the cation exchange capacity (CEC) of bentonite.

We comparatively conducted radionuclide sorption tests for the initial and altered bentonites. We measured their sorption amounts and investigated their sorption difference.

Keywords: Bentonite, Montmorillonite, Indigenous bacteria, Hematite, Reduction

Reviews of the Study on the Alternative Options for High Level Radioactive Waste Management

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The spent nuclear fuels from domestic nuclear power plants can be classified as high-level wastes(HLW), so they are disposed of directly or considered with a strategy to dispose of radioactive waste generated from the recycling process of spent nuclear fuel for useful substances. Whether considering direct disposal of spent nuclear fuel or disposal of high-level waste generated from the recycling process, it is essential to keep the public safe and to maintain isolation from the human environment for a long time of tens of thousands of years.

The international scientific consensus has been concluded that deep geological disposal is the best method for the disposal of HLWs with current technology. A deep geological disposal system is designed to contain nuclear waste several hundred metres or more below the surface, in a stable geological site (such as granite, clay or salt formations) that isolates the waste from the biosphere, thus preventing contact with humans and the environment. The KBS-3 concept that is a kind of deep geological disposal system developed by SKB, Swedish radioactive waste management company, is currently considered as the reference concept with current technology. In Korea, also this KBS-3 type concept has been considered as a reference concept for geological disposal of spent nuclear fuels or HLW.

But, because the project for high level radioactive waste management takes a long time period, a flexible approach is needed and the related research and development of alternative geological disposal options to support optimized implementation has been carried out essentially in the advanced countries in the area of HLW management. So monitoring of such research and developments on alternative options to the reference geological disposal concepts for HLW as part of ensuring safer, more effective disposal and flexibility in the countries is needed.

In this study, the research and developments on the alternative geological disposal options for high level radioactive wastes including spent nuclear fuels from nuclear power plants in advanced countries of HLW management for flexibility or disposal efficiency, such as Sweden, USA, Canada, UK, were reviewed. The result of review for geological disposal option studies can be used in establishing the direction for research and development on alternative options of geological disposal.

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Keywords: Spent nuclear fuels, High-level waste, Geological disposal, Alternative options, Flexibility

The Role of Natural Analogue in Spent Fuel Disposal Research

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It is imminent task to deal with spent fuel, since the interim storage on site is getting saturated in Korea. In addition, it takes a quite long time to prepare the systematic way to deal with spent fuel disposal, because it requires minimize the uncertainty of technical application and social disagreement.

A natural analogue is a natural historical or anthropogenic system that permits a study of repository-related processes including its surrounding environment and the processes that control its evolution. Natural analogue has advantages such as directly observable in the environment and offer links to personal experience to understand easily, but disadvantages including complex integrated environmental processes thus hesitancy in their use by experimental scientists, modelers and physical scientists.

In this study, we are to review the previous studies as follows: relevance, limitations, quantitative information, time scale, safety case applications and so on. In this review, we are to discuss what makes a good or bad analogue study, so that we could use the insight to initiate our new program and choose study sites.

Keywords: Natural analogue, Spent fuel disposal, Limitation, Advantage

Quantitative Analysis of the Hydrogeological Change of Rock Mass in the Vicinity of Borehole Surface

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The changes in the permeability of a rock mass due to the expansion of diameter of the borehole were investigated, which was considered as an excavation of a disposal hole in this study. After drilling a borehole with a diameter of three inches, in situ hydraulic tests were carried out in 12 sections at 1.45 m intervals. After the borehole was expanded to four inches, the permeability of the rock mass were determined by comparing them with those derived from the same test section before the borehole expansion. The mean value of the transmissivity derived from the hydraulic test before the borehole expansion was 1.5E-08 m^2/s, which was, about four times the average transmissivity of 5.5E-08 m^2/s obtained after the borehole expansion. From the results of hydraulic tests, we could expect the increase in permeability in the vicinity of disposal hole after excavation, which was known as the excavation damaged zone (EDZ) in the field of rock mechanical engineering. During the analyzing the pressure pulse test, the equivalent radius derived from the pressure pulse test was increased after borehole expansion. Furthermore, the increase in the equivalent radius after the borehole expansion was similar to the change in the transmissivity of the test section based on the pressure pulse and slug tests. In the test section in which the equivalent radius increased at least 5 times or more, the transmissivity of the test section also significantly increased compared with that of other test sections. In addition, the ratio of the transmissivity from steady-state analysis and the transient-state analysis from the constant head test before and after expansion was decreased. In this paper, to quantify the changes in the hydrogeological properties due to the borehole expansion, the differences between the equivalence radii derived from the pressure pulse test and the results of the constant head test were evaluated. The two results could not be compared because the constant head test was not performed in the test section in which an equivalent radius significantly increased. The results of the constant head test indicated that the transmissivity of the medium changed due to the expansion of the boundary between the two-dimensional and three-dimensional flow.

Keywords: KURT, Borehole expansion, Pressure pulse test, Slug test, Constant head test, Equivalent radius

Review of Degradation of the Spent Nuclear Fuel Matrix for a Safety Assessment

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Chemical stability of solid waste matrix is a key safety concept of a geological disposal system for spent nuclear fuels (SNFs). This study discusses features, events, processes (FEPs), and modeling approaches relating to the release of radionuclides bounded in the UO_2 matrix. Based on the FEPs analysis, critical parameters influencing the dissolution of the UO_2 matrix are derived. These include a concentration of carbonate, production of hydroperoxide by alpha radiolysis of groundwater, and production of hydrogen by the corrosion of a disposal canister. From this review, we concluded that both the anodic and the chemical dissolution of the UO_2 should be taken into account.

Based on the FEPs analysis, the dissolution rate of the UO_2 matrix is calculated. We considered two modeling approaches, including an insight modeling method and mixed potential theory. In the insight modeling approach, the complexity relating to the long-term degradation of the waste matrix is simplified. Three mathematical models in simple geometrical systems were introduced and compared to each other: solubility limited dissolution model, kinetics and mass transfer limited model, and mass transfer resistance model. The dissolution rates estimated by the solubility limited dissolution model were the most conservative (10^{-9}~10^{-8} fraction/yr).

The insight models do not simulate the chemical and the anodic dissolution reactions relating to the degradation of the UO_2 matrix. In the mixed potential model, the anodic dissolution of UO_2 is estimated considering critical redox reactions. It is concluded that the anodic dissolution of the UO_2 by the alpha radiolysis of groundwater cannot be ruled out during thousands of years. For this reason, a fractional dissolution rate of 10^{-7} fraction/yr would be reasonable for the safety assessment of the KAERI reference disposal system. In a future study, however, input parameters for the mixed potential model should be carefully reviewed because lots of data are based on assumptions yet.

Keywords: Spent nuclear fuel, Matrix degradation, Radionuclide release rate, Disposal system

Modelling Radionuclide Transport Through Biosphere Using Modified SWAT

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The biosphere is the environment where living organisms inhabit, including the soil, surface water body such as rivers, lakes, and seas, and the aquifer. Geological disposal system shall be designed to isolate radionuclide from the biosphere over hundreds of thousands of years. However, radionuclide may be released from the repository and transported to the biosphere, if any hypothetical releases occur possibly due to inadvertent human intrusion or canister failure. Therefore, it is important to model the transport of radionuclide in the biosphere to assess the potential radiation exposure of human and other living organisms existing in the biosphere. For this purpose, this study modified SWAT (Soil and Water Assessment Tool), the watershed-scale hydrological model developed by the USDA-ARS (United States Department of Agriculture – Agricultural Research Service), in order to simulate the radionuclide transport through biosphere. The SWAT delineates a watershed into HRUs (Hydraulic Response Unit) and simulate water cycle and contaminant transport in the watershed consisting of multiple HRUs with different soil and land use types. In this study, the SWAT was modified with two features: 1) the DHRU (Disaggregated HRU) transformation methodology used in the SWAT-MODFLOW model was applied, in order to link the result of the FEM grid-based geosphere model (i.e., APro; Advanced Process-based total system performance assessment framework for a geological disposal system) with the HRU of SWAT; and 2) decay chains of radionuclides were applied. For the test of the modified SWAT, the model was built at the watershed around the KURT site and run for 1,000 years in a daily basis. During the simulation period, the surface environment was assumed constant. Additionally, constant release fluxes of several radionuclides from the geosphere-biosphere interface (GBI) was applied based on the reference values of complementary safety indicator from IAEA literature. As a result, the modified SWAT showed the time-series concentration of radionuclides in each watershed component. This model will be a useful tool for the safety assessment of a geological disposal system. However, the model needs to be improved for the more accurate quantitative analysis, since the SWAT model itself contains the uncertainty due to the simplified groundwater flow module.

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

Correlation Analysis of Hydraulic Conductivity and Geophysical Characteristics

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In this study, we analyzed the correlation between the hydraulic conductivity and geophysical logging results as a method for understanding the permeability of the crystalline rock. Plus, we would like to find out which geophysical characteristic explained well the permeability of the medium. KP-3 borehole, which was drilled at KURT, was used for hydraulic tests and geophysical logging. To determine the hydraulic conductivity of crystalline rock, we conducted a constant head withdrawal test (CHWT). Also, the geophysical characteristics and fracture frequency were obtained by using natural gamma logging, resistivity logging, and drilling core logging in the KP-3 borehole. The geophysical investigated results were divided according to the hydraulic test sections of CHWT, and representative values were inputted for the value of each section. These representative values were used to obtain the correlation coefficient with hydraulic conductivity, and factor analysis (FA), and principal component analysis (PCA) was also performed. As a result of analyzing the correlation between the hydraulic conductivity and the geophysical characteristics measured in the KP-3 borehole, a strong correlation of 0.83 was shown with the variable "Fracture (open)". That was, the frequency of fracture explained permeability well. With Rock Quality Designation (RQD), which was a variable with a negative correlation, the correlation of -0.67 was shown. Also, in FA, the variables that correlate hydraulic conductivity and characteristics showed the same distribution as in the correlation analysis. In the PCA results, the variables most related to hydraulic conductivity was “Fracture (open)” and mostly unrelated to hydraulic conductivity was RQD.

Keywords: Constant head withdrawal test, Geophysical characteristic, Correlation, Factor analysis, Principal component analysis

Recent Technologies in the Concept, Manufacturing, and Installation of Backfill for Disposal Tunnel

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In high-level repositories, backfill is one of the important components of an engineered barrier and is a barrier material that fills the deposal tunnel. The major function of the backfill is to act as a hydraulic and mechanical barrier: it blocks the flow of groundwater in the deposal tunnel so that the released radionuclide is dominated by diffusion rather than by advection transport; it prevents the bentonite buffer in the deposition borehole from rising in the direction of the upper deposal tunnel by swelling due to groundwater inflow; the excavated deposal tunnel maintains structural stability against external loads. The purpose of this paper is to review the concept, manufacturing, and installation technology status of backfill for disposal tunnel reported in the literature, and present recent technology applicable to an HLW repository in Korea.

Seven backfilling techniques have been studied so far: compaction of a mixture of bentonite and crushed rock (or sand) in the tunnel; compaction of natural clay with swelling ability in the tunnel; compaction of non-swelling clay in the tunnel combined with application of pre-compacted bentonite blocks at the roof; placement of pre-compacted blocks in the entire tunnel cross-section; sandwich concept and compartment concept techniques which consist of crushed rock compacted in the tunnel and pre-compacted bentonite blocks; three step-backfilling technique of levelling of the tunnel floor with pellets, placement of pre-compacted blocks, and introduction of pellets to fill the space between the blocks and the tunnel walls/ceiling. Of these seven backfilling techniques, the last one is the most preferred, and Finland, Sweden, and Canada have decided to apply this technique to backfilling their disposal tunnels.

Pre-compacted blocks, in the three-step backfilling technique, were prepared using bentonite-crushed rock mixture, and pellets were prepared using pure bentonite. Backfill blocks were manufactured using the uniaxial compaction method, and bentonite pellets were manufactured using the roller compression method. The installation of the backfill blocks and pellets were as follows: leveling of the tunnel floor with bentonite pellets was done by in-situ uniaxial compaction; the blocks in the tunnel and pellets between the blocks and the tunnel walls/ceiling were installed by a program-automated method using the same installation device.

In summary, the three-step backfilling technique is a good alternative applicable to our repository in terms of the safety function of the backfill. However, Korea is in the early stage of research on backfill, and even its concept has not been developed yet. Systematic research on the backfill is urgent, and based on the three-step backfilling technique, it is necessary to select components and materials of backfill suitable for the concept of Korean disposal system, and to develop manufacturing and installation technologies.

Acknowledgements

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Keywords: Backfill, Disposal tunnel, Engineered barrier, High-level repository

Development of a Web Application for 3D Underwater Scan Calibration in Nuclear Decommissioning

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For highly contaminated elements such as reactor pressure vessels or reactor internals, it is a viable option to cool-down and dismantle these elements in underwater environment. However, underwater 3D scanning faces technical challenge; 3D data scanned underwater has distortions and requires calibration to implement an accurate working environment. This study introduces a microservice-style web application for underwater 3d scan data calibration. Calibration was carried out using deep learning models. Deploying the calibration model as a microservice on a cloud application platform makes users easily use the model without having to build a separate environment. Cloud application platforms such as Heroku also provide visualization tools, allowing users to immediately see calibrated outputs.

Keywords: Web application, Deployment, Micro service, Deep learning

Preliminary Study on the Lithological Characteristics of Jurassic Granitoids for Construction of Geological Evolution History Around KURT 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 assessment of the repository, the long-term natural evolution and possible events of the site, that can cause disturbances to the facility over the period of interest, should be considered. Studying the geological development processes that the site has experienced can contribute to understanding and describing the present-day conditions. Moreover, knowledge of the past is necessary to predict the future evolution of the site. To describe this long-term geological evolution, information from the site investigation is needed in addition to the scientific literature survey. The database for analyzing geological development processes can be represented as an integrated model, covering the current state of the geological environment and natural processes that affect long-term safety. It will also be comprised of a description from the site investigation and regional setting. To construct the methodology for the development of a long-term geological scenario and a final safety case report, we have been investigating the area around KURT as a test site, which is the underground facility located in a massive Jurassic igneous body. In this research, we studied the difference in the petrological characteristics of deep underground igneous granitoids using boreholed rock-core and sampled rocks from the KURT. This is a preliminary study for the future construction of a lithological and integrated model. The results can also serve as part of a database to establish a criterion for the classification of lithology comprised mainly of igneous granitoids and their evolutional history.

Keywords: Geological evolution, Radioactive waste disposal, Lithological model, Long-term safety

Benchmark Simulation of Radionuclide Transport in a Fractured Rock Using Explicit and Implicit Fracture Models

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In order to ensure the reliability of the safety of the geological disposal system, it is essential to evaluate the performance of multi-barrier systems. For this purpose, the concept of a process-based performance assessment model for a geological disposal system called APro (Advanced Process-based total system performance assessment framework for a geological disposal system) was proposed by Korea Atomic Energy Research Institute (KAERI). In order to build confidence and to improve the models and methodologies of APro, KAERI is participating DECOVALEX-2023 Task F which is the international joint program for the comparison of the models and methods used in post-closure performance assessment of deep geological repositories. Task F have 4 stages which focuses on the comparison of forward models with increasing complexity: Benchmarks of process models, Deterministic Reference Case, Uncertainty Propagation, and Sensitivity Analysis, and in this study, two benchmark simulations for the transport of radionuclide in a fractured rock were conducted as a part of Task F. For the benchmarks, particle tracing tests were carried out numerically on rock masses with 4 deterministic fractures and stochastic fractures. Since COMSOL Multiphysics, which is responsible for simulating fluid flow and radionuclide transport at APro, is a continuum code based on Finite Element Method (FEM), however, numerical techniques are needed to simulate fractures using a continuum model. For this reason, two methodologies were applied: a methodology that simulates the fluid flow in the internal boundary using Discrete Fracture Network (DFN), and a methodology that simulates the fluid flow in a rock mass by upscaling with Equivalent Continuous Porous Medium (ECPM). With these two numerical techniques, the applicability of numerical model was evaluated through the breakthrough curve, and also the effects of decay and sorption were evaluated considering the half-life and the retardation factor, respectively.

Keywords: DECOVALEX-2023 Task F, APro, Radionuclide transport in a fractured rock, Discrete fracture network, Equivalent continuous porous medium

4분과

중저준위폐기물관리 (Oral)

Preliminary Safety Assessment of VLLW Disposal in the Well Scenario Using RESRAD-OFFSITE

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Starting with the permanent shutdown of the Kori-1 and Wolseong-1 reactors, immediate decommissioning of nuclear power plants is planned in Korea. As a result, radioactive wastes are expected to be generated in large quantities, and then about 67% of them would be classified as VLLW (Very Low Level Waste). Therefore, a trench-type VLLW disposal facility has to be constructed timely in accordance with the decommissioning work plan of Kori-1. In the VLLW disposal facility construction project, post-closure safety assessment plays an essential role, and it is necessary to utilize appropriate computer simulation codes for this purpose. A preliminary assessment was in this study, conducted to investigate the applicability of RESRAD-OFFSITE to the safety assessment of VLLW disposal facilities. The well scenario in groundwater release pathway was applied to consider exposure to drinking water using wells after the closure of disposal facility.

In RESRAD-OFFSITE, four options are available for conceptualization of releases from primary contamination i.e., releases of radionuclides from waste materials to the surrounding solids. The "Specific Initial Activity Base on Mass of Entry Primary Contamination" option was applied in this study to consider contamination of all solid wastes. In the case of VLLWs, most of them can occur in various forms, such as sludges, small metals, and concrete wastes. Considering the diversity of waste types, the Equilibrium Desorption Transfer mechanism and First Order Rate Controlled Transfer mechanism of radionuclide-specific release to groundwater in RESRAD-OFFSITE were applied to assess the source term. The input data were set by referring to the previous study using GoldSim and the maximum calculation time was set to 100,000 years.

Comparing the annual individual peak doses from use of well water for the major radionuclides (C-14, H-3, I-129, Nb-94, Ni-59, Tc-99, and gross-alpha), it can be seen that there is a good agreement (within and order of magnitude) between GoldSim and RESRAD-OFFSITE.

From this preliminary assessment, it is found the RESRAD-OFFSITE could be appropriately applied to post-closure safety assessment for the VLLW disposal facility. Taking easy-to-use advantage and considering various source-term modeling capability, it is thought RESRAD-OFFSITE would an useful tool for comparison of results with other codes.

Keywords: RESRAD-OFFSITE, Safety Assessment, Very Low-Level Waste, Disposal Facilities

A Study on the Technology to Secure Disposal Safety by Sintering of Pellets Compressed With Dried Powder of Concentrated Liquid Radioactive Waste in NPPs

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In order to disposal of dried fine powder of concentrated liquid radioactive waste in NPPs, it must be treated and packaged so that it is non-dispersive. In this study, a technology for securing disposal safety to solidify the pellets was reviewed after manufacturing the dried powder of NPPs concentrated liquid radioactive waste into a pellet with a constant particle size and hard. A simulated sample having a similar composition to the concentrated liquid radioactive waste of NPP was prepared, and dried using the CWDTS (Concentrated Waste Drying Treatment System) developed by our company (LC Gen Co., Ltd.) to produce powder. And then, the dried powder was compressed into pellets with a size of about 8 mm by compression molding with a compression molding equipment, in order to increase the compressive strength of these pellets, sintering was performed using a sintering equipment. Since the main component of the concentrated liquid radioactive waste dried powder is boron, which is a metalloid, the microwave sintering method was selected as a sintering method in consideration of sintering and fire protection. The microwave sintering equipment developed in this study uses 2,450 MHz high frequency PID (Proportional Integral Differential) control and micro time slicing temperature control (milliseconds) to maintain a constant sintering temperature and promote densification by self-heating by three-dimensional irradiation. The sintering test was sintered at a temperature of 100℃ for 15 minutes, and the standard value of the friability test of the pellets after sintering was within 1%, but the test result was 0.1%, which was more satisfactory than the standard value, confirming that the surface was much harder. In addition, in the compressive strength test, the compressive strength standard of soft solidification is 4.22 kgf/cm^2 or more, but it was confirmed that the test result was very high as 27.2 kgf/cm^2. As such, it is believed that the safety of disposal will be secured if dried powder of concentrated liquid radioactive waste in NPPs is manufactured into pellet and then sintered it.

Keywords: Concentrated Liquid Radioactive Waste, Dried Powder, Disposal Safety, Pellet Sintering, CWDTS, Sintering Equipment

Chemical Affinity Quantum Sieving Effect of Fluorine for Isotopic Separation of Tritium From Radioactive Wastewater

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The separation of radioactive tritium from radioactive wastewater presents a new challenge for the nuclear and fusion industry. The most promising methods are based on isotopic deviations of the zero-point energy in chemical bonding incorporating hydrogen isotopes. This isotope effect can be translated into different formation enthalpies between proton and tritium at isotope discriminative chemically active sites in solid materials, which is termed Chemical Affinity Quantum Sieving.

This research provides a theoretical and experimental demonstration of the chemical affinity quantum sieving effect of fluorine for tritium separation. For the theoretical approach, tritium to protium isotopic reduced partition function (IRPF; *s/s'f*(T/H)) was calculated from previous research data. From this calculation, fluorine ($IRPF_{fluorine}$=28.496) is expected to form the highest isotopic differential chemical bonding with hydrogen. Therefore, fluorine was used as the best potential candidate for surface active sites which can be used for chemical isotope separation of tritium.

For the experiment, fluorinated MCM-41 (F-MCM-41) was prepared by thermal treatment with different amounts (N=0.2 g/ 0.5 g/ 1.0 g) of solid NH_4F in 60 mL of isopropanol at 120°C for 24h. Tritium separation factor of pristine MCM-41 and (N)F-MCM-41 was comparatively tested with batch sorption test with tritiated water (10,000 Bq/mL) at 6°C. The F-MCM-41 showed significantly enhanced tritium isotope separation factor(α) over pristine MCM-41, indicating the formation of isotope discriminative chemical active sites by fluorination ($\alpha_{pristine/6°C}$=1.20, $\alpha_{0.2F/6°C}$=1.56, $\alpha_{0.5F/6°C}$=3.29, $\alpha_{1.0F/6°C}$=1.59). F-MCM-41 synthesized with 0.5g of NH_4F showed an optimized isotope separation factor (α=3.29), which is comparable to the chemical isotope exchange reaction of the CECE process (α=3.7).

In order to elucidate the mechanism of isotope discriminative removal of tritium, solid state ^{19}F NMR spectra were obtained for each sample before and after sorption reaction. After sorption, all the tested samples showed a new octahedral chemical environment, which can be assignable to $H_2SiF_6{\cdot}4H_2O$ crystal. The intensity and integrated signal area of this $H_2SiF_6{\cdot}4H_2O$ peak showed a good correspondence with tritium isotope separation factor of (N)F-MCM-41s. Therefore, we conclude that the formation of $H_2SiF_6{\cdot}4H_2O$ in MCM-41 framework is the main contributor to tritium separation from radioactive wastewater, and SiF_6^{2-} is a chemical active site for tritium removal.

This research sheds light on FLUORINE as a chemical active site for hydrogen isotope differential sorption, and suggests a direction for the future design of novel material for the selective removal of tritium from radioactive wastewater. In addition, this research is expected to be used for establishing the basis of the new tritium separation process using solid-liquid interfacial reaction.

Keywords: Tritium, Isotope separation, Chemical quantum sieving, Fluorine, Isotopic reduced partition function

Development of the Pyrolysis for Volume Reduction of Organic Solid Waste Containing Uranium

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As a treatment method for the final disposal of general solid waste, the management of waste is comparatively convenient with incineration, landfill, recycling, etc. However, since the management of radioactive waste is strictly applied, development of appropriate treatment process is necessary. Organic solid radioactive waste is produced from nuclear power plant, nuclear research centers, radiopharmaceuticals and nuclear fuel fabrication plants, etc. Physicochemical treatment techniques such as cement solidification and incineration have been applied to treat the solid waste.

Incineration technology is one of the stable heat treatment techniques that have been proven for a long time. The incineration technology has the advantage of the largest volume reduction rate of waste. But because gaseous pollutants are generated at a higher temperature, amount of pollutants which is emitted at high temperatures such as dioxin and NOx is higher than other heat treatment methods. On the other hand, pyrolysis technology doesn't generate exhaust gas, dioxin, and NOx compared with incineration technology, because pyrolysis is reacted at low temperature. It is expected that the amount of uranium particles from syngas of pyrolysis is also small due to less scattering of heavy metal substances compared to incineration technology.

In this study, the result of pyrolysis process was investigated for the volumetric effect of uranium - containing organic solid waste generated in a nuclear fuel processing facility.

To reduce the volume of organic solid waste in nuclear fuel fabrication facilities, a pyrolysis process was developed. The volumetric reduction of the solid waste was shown to be up to 87% when this process was applied. It was confirmed that the treatment was possible.

The results of this study suggest that the development of the pyrolysis process is the optimal treatment method to achieve homogenization of waste in the treatment of organic solid waste generated in nuclear fuel fabrication facilities and satisfy both stability and waste reduction.

However, additional char treatment facilities will be needed for final disposal, and the research is expected to be carried out later.

Keywords: Pyrolysis, Organic Solid Waste, Volume Reduction, Uranium

Development of Calculation Program for Density Correction Efficiency of Gamma Spectroscopy

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This study is to develop a program to calculate the density corrected efficiency to be applied to waste characterization whose density is not same as the density of the standard source. The determination of the density correction factor was not easy and laborious. so was carried out only in specialized analysis Laboratories, It has been often ignored in nuclide analysis fields. there was a high systematic uncertainty because density correction was not performed in the nuclide analysis of radioactive waste whose density was not 1 g/cc.

The density correction factor calculation program developed in this study is based on the measured efficiency from 1 g/cc of the source. The density corrected factor is calculated by a correction factor according to the difference in density between the sample and the 1g/cc standard source. The measured efficiency reflects the long-term degradation and the characteristics of the detector. The density correction factor under a specific container and geometry is determined by MCNP code. The program from To determine the density correction factor, six certified reference material with densities ranging from 0.02 to 5.44 g/cc were measured. Each source emits multiple energy lines ranging from 88 to 1836 keV. The correction factor was calculated using the log fitting curve function according to the density of the measured standard Source. The density correction calculation program is automatically calculate after waste weight input.

The corrected efficiency is automatically applied to Genie-2000 of Mirion (Canberra) and GammaVision of AMETEKORTEC, the analysis program in use. The program can run under the Window 10 64 bit and Microsoft .NET Framework 4.5. The program is available for a relative 40% coaxial HPGe detector, 1 L cylindrical beaker and 1 L marinelli beaker, measurement geometry of 5 mm away from detector surface, energy range from 50 to 2,000 keV, density range from 0 to 7.00 g/cc of density range.

The developed program allows more accurate nuclide evaluation of solid wastes having various densities. Therefore, it is possible to perform more reliable evaluation of nuclides compared to conventional analysis methods. This means that the radionuclide concentration can be accurately assessed, which allows safer clearance to be performed.

Keywords: Monte-carlo Simulation, Density Correction Factor, Density Corrected Efficiency, Clearance Waste

Evaluation of Stacking Deformation Inside Disposal Facility According to the Void-filling Rate of the NPP Decommissioning Radioactive Waste Disposal Container

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It is a well-known fact that a huge amount of radioactive waste is generated by decommissioning nuclear power plants. To efficiently dispose of such decommissioning radioactive waste, the development of various low and intermediate-level waste disposal containers has been actively progressing. In line with this movement, it is necessary to develop a disposal container specialized for low and intermediate-level waste.

Radioactive waste containers must be able to withstand a load equivalent to 5 times the container weight following related laws. Therefore, the radioactive waste container has basic structural performance. However, the generally applied 85% void-fill rate condition may cause vertical deformation of the non-structural element when an excessive load is applied to the top of the container. Therefore, there is a method that can control the vertical deformation according to the container void-filling rate condition.

In this study, the basic method for preventing the deformation of the disposal container used for decommissioning radioactive waste was analyzed. As for the disposal container, a large disposal container that can be used in a trench-type disposal facility was selected. The effect of deformation according to the container void-filling rate on the disposal container stacking environment was analyzed.

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Keywords: Decommissioning radioactive waste, LILW, Disposal container, Void filling condition

A Preliminary Study on Feasibility of Identification of Gamma Emitting Radionuclides Using Mass Attenuation Coefficient of Shielding Materials

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After permanent shutdown of nuclear power plant, lots of radioactive wastes are generated during the process of decommissioning. IAEA emphasizes that testing and analyses to demonstrate the radioactive content and the quality of final waste forms and waste packages are key components of control and are essential to accurate characterization of the waste. Also, IAEA GSR Part 5, Radioactive waste has to be characterized in terms of radiological properties. The relevant characteristics of the waste have to be recorded to facilitate its further management. In this research, We tried to suggest the way to figure out the composition of gamma-emitting radionuclides. Particularly, ^{137}Cs, ^{60}Co. They are key nuclides in radioactive wastes and can be readily measured by non-destructive assay means. Linear attenuation coefficient(μ) is a constant that describes the fraction of attenuated incident photons in a material. This coefficient increases with increasing atomic number and density of the absorbing material and decreases with increasing photon energy. Generally, ^{137}Cs emits 0.662 MeV(85.1%) and ^{60}Co emits 1.17(99.8%) and 1.33 MeV(99.9%) γ-rays per disintegration. 0.662 MeV γ-ray must be more attenuated than 1.17 and 1.33 MeV γ-rays.

Using equivalent dose rate equation and conversion coefficient, we calculated ambient dose equivalent(H*(10)) which is operational quantity. In this step, we identified ambient equivalent dose rate decreased much more when a source included more radioactivity of ^{137}Cs than ^{60}Co. This phenomenon occurs because 0.662 MeV γ-ray attenuated much more when it penetrates material such as lead and copper. Degree of decrease was different according to absorber's thickness, type and composition of sources. Using this property, we measured decreased ambient dose equivalent rate(μsv/h) by using RDS-31 survey meter, lead, copper absorbers and disk sources(^{137}Cs, ^{60}Co). As a result, A source which included radioactivity of ^{137}Cs more than ^{60}Co represents more attenuated ambient dose equivalent rate. Furthermore, we roughly estimate their ratio of radioactivity by analyzing the degree of dose rate's decrease, and by manipulating absorber thickness and radioactivity of ^{137}Cs and ^{60} Co by changing composition of sources. For the validation, A source which has 67.9% radioactivity of ^{60}Co was measured to have 57.1% and 61.9% radioactivity of ^{60}Co when using copper and lead absorber each.

This study will contribute to simple and roughly In-situ measurement which can characterize the most representative gamma-emitting radionuclides. Also, this can be used to support characterizing other radionuclides remaining in radioactive wastes.

Keywords: Linear attenuation coefficient, Composition, Ambient dose equivalent, Radioactive waste, Characterize

Study on the Basic Properties of Sintering for Treatment of Uranium Contaminated Soil

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The study of volume reduction and immobilization of radioactive waste should continue, due to its high disposal costs and strict disposal regulation. In addition, with the imminent dismantling of nuclear power plants, the waste from decommissioning should be efficiently reduced as this will result in many types and large amounts of radioactive waste.

Among various types of radioactive wastes, the contaminated soil might account for a large portion. In this study, the experiment using simulated soil sample was conducted as a preliminary test of volume reduction and immobilization of uranium contaminated soil that can be generated from a related nuclear facility. In accordance with the radioactive waste disposal regulations, it is necessary to immobilize the removed soil from the site because the waste to be disposed should not have fluidity and dispersibility. We have carried out the evaluation of the basic properties for soil sintering technology that can simultaneously employ the volume reduction and stabilization preparing for the large quantities of soil waste.

The soil texture, moisture content, organic content, pH, and bulk density of the soil sample were measured, and XRD, XRF, and TG analyses were used to determine the physicochemical and mineralogical properties of the soil. The soil was divided into sand (0.005 ~ 2 mm) and fines (< 0.005 mm), and each fraction was pelletized with high pressure (> 100 MPa) as a 13 mm of diameter and then sintered using a muffle furnace at high temperatures (>800℃). As a result of measuring compressive strength of the sintered pellet, it was confirmed that the sintered soil has stabilized sufficiently. Also, the soil sintering method was effective in reducing the final volume of disposal waste.

Keywords: Radioactive waste, Contaminated soil, Sintering, Volume reduction, Immobilization

4분과

중저준위폐기물관리 (Poster)

Compressive Strength Analysis of Polymer–Cement Along With Their Mixing Ratio as a Solidifying Agent for Radioactive Waste

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A project on the development of high performance cement based materials and solidification technology for immobilization of liquid radioactive waste from decommissioning of nuclear power plants is being conducted. For this a polymer-cement is being developed as a solidifying agent. An epoxy resin was selected for the polymer because its viscosity is comparatively lower than other polymer resins. Generally, epoxy is emulsified, but emulsification gives a tendency to require longer time for curing of the polymer-cement, and to cause shrinkage of the mixture. To reduce the curing time and shrinkage, non-emulsified epoxy with hardener, i.e. Radsol 1000, was used. For the cement, Portland cement was introduced.

To find out a suitable mixing ratio of polymer-cement, compressive strength was measured with varying ratios, i.e. 4.5:0, 4.5:1, 4.5:2, 4.5:3, and 4.5:4. The compressive strength was found slowly increased with the increase of the cement ratio. For all mixing ratios tested, measured strengths were over 8,000 psi (criterion for hard compressive strength: 500 psi). The polymer-cement with the ratio of 4.5:4 was very sticky and, therefore, expected to cause difficulty in its injection into the dried waste product by vacuum. Accordingly, this case was excluded in the ratio analysis. With the remaining ratios, the most suitable ratio of polymer-cement was found to be 4.5:2 in the aspect of compressive strength, which could allow a favorable portion of the dried waste. And also, the compressive strength was measured along with curing time of this polymer-cement. The strength was very slowly increased up to 14 days with the increase of the curing time, before being stabilized. This suggests that about 14 days might be reasonable for its curing, reaching an average of 13,100 psi, far exceeding the acceptance criterion for compressive strength. In terms of compressive strength, the mixing ratio of 4.5:2 for non-emulsified epoxy-cement as a solidifying agent was found appropriate, and 14 days could be sufficient for curing the polymer-cement.

However, further study should be undertaken to confirm the suitability for disposal of the final waste form of liquid waste solidified by the polymer-cement with its mixing ratio of 4.5:2.

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Keywords: Polymer cement, Compressive strength, Mixing ratio, Curing time

A Study on the Manufacturing of Polymer Waste Form Incorporated Pellets Using the Polymer Solidification System

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Powdered waste evaluated as disposal nonconformity radioactive waste must be packaged and treated to be disposed non-dispersibility. At this time, these powdered waste can be solidified with solidification agent such as cement and polymer. However, the existing method of cement solidification, each particle should be enclosed by solidification agent so, the volume of final disposed waste form will be increased. Therefore, in this study a method was devised to form simulated sample, into high-density pellets using roll compactor. Then, these pellets were filled maximum in drum and polymer was injected to the void between pellets. This will not increase volume of the final disposed waste form and rather reduce the disposal costs.

The purpose of this experiment is deducted the optimal curing conditions. Therefore, it was measured that viscosity control of epoxy polymer, possibility of crack according to curing temperature and compressive strength of specimens. The results of viscosity measurement for four types of epoxy according to the injection of diluent (LGE), it was confirmed that main ingredient (YD-128) and hardener (G-1034) are easy to adjust viscosity even with a small amount of LGE. The heating temperature was measured, when the polymer curing, to confirm that curing time was shorter at 50℃ and the possibility of the specimen cracking was reduced. Furthermore, the pre-compressive strength of specimen which curing at 50^{o}C was the highest at 25.11 MPa. Therefore, the mixing ratio of epoxy polymer was YD-128:G-1034 being 65:35 wt.%, and LGE was added 10 wt.% of the total mixture. The incorporation ratio of the pellets was higher than 60 wt.%, and waste forms were cured in the thermostat capable of maintaining the temperature at 60^{o}C.

In consideration of the above conditions, a polymer solidification system was deducted. Under the same conditions, polymer waste form was manufactured and then characteristic evaluations (compressive strength, thermal cycling, immersion and irradiation tests) were performed on these specimens. The results of measuring compressive strength after each test, were 24.80 to 29.20 MPa, which is exceeded Waste Acceptance Criteria (WAC) for disposal facility in Korea. Especially, after irradiation test, compressive strength was the highest at 29.20 MPa, which was judged to have the best structural integrity. Finally, the Volume Reduction Factor (VRF) was used to verify volume reduction ratio of specimens. VRF was 4.26, meaning that polymer waste forms incorporated with pellets could be compressed to 1/4.26 times.

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Keywords: Powdered waste, Pelletization, Solidification, Polymer waste form, Volume reduction

Development of High Efficient Compactor for High Dose Spent Filter

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Much high dose spent filters were produced at nuclear power plants since 1983 and from 1983 to 1994 they were solidified with cement, and from 1995 to 2006 they were simply packed in drums, and since then they were stored at storage areas at nuclear power plants for attenuating dose. High dose spent filters are needed to be compacted for securing the disposal compatibility for cave disposal at Gyeongju disposal site in consideration of the decommissioning of Kori #1 plant, and the timely treatment of spent filters are required before the start of decommissioning.

For compatibility of disposal the compactor having super compaction capability is required. In this regard Orion E&C developed the new idea design of high efficient compactor for spent filters by adopting 3 steps of compaction by x, y, z direction.

The compaction forces are 150 tons, 250 tons and 350 tons by x, y, z direction respectively and this 3 steps compaction can reduce the volume of spent filter at more than 8 reduction ratio (actual ratio is calculated at 8.13). The force per unit surface area to transferred to spent filter at z direction reaches to 3,000～4,000 tons.

The compactor size is 1.8m(L) x 1.4m(W) x 2m(H) and the weight is about 4.5 tons, and this needs 8 minutes per one cycle.

This compactor will give high efficient capability for spent filter compaction and economical benefits by reducing the volume as well as the disposal cost (15.19 million won per 200 L drum).

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Keywords: Compactor, Spent filter, Disposal compatibility, Reduction ratio, Super compaction

Performance Evaluation of the Daily Waste Throughput of a Radioactivity Measurement System for Standard/Non-standard Radioactive Waste

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In this study, our research team developed a radioactivity measurement system for standard/non-standard radioactive waste consisting of a shape measurement equipment, a PVT (polyvinyl toluene) screening equipment, a HPGe (high purity germanium) scanning equipment, and a classification equipment and then we carried out the performance evaluation of daily waste throughput. After the radioactive wastes were sequentially transferred to the shape measurement equipment, the PVT screening equipment, and the HPGe scanning equipment, the time they were classified and discharged by the classification equipment was measured through experiments. Here, the weight of the waste stored in the 200 L drum was obtained by randomly calculating the density and filling rate of the waste and multiplying the volume of the drum. Experimental results on waste throughput, 55 seconds for shape measurement equipment (history acquisition, metal/non-metallic discrimination, weight measurement, shape acquisition and transfer), 355 seconds for PVT screening equipment (shielding door opening and closing, screening and transfer), 1,980 seconds for HPGe scanning equipment (shielding door opening and closing, scanning and transfer), 57 seconds for classification system (rotation, transfer and discharge) were measured with the average processing time per case on each equipment. Accordingly, if contamination is determined by the PVT screening equipment, the total processing time was measured at the shortest time of 467 seconds. If non-contamination is determined in the PVT screening equipment and non-contamination in the HPGe scanning equipment, the total processing time was 2,390 seconds. Finally, if non-contamination is determined in the PVT screening equipment and contamination in the HPGe scanning equipment, the total processing time was measured at the longest time of 2,390 seconds. Conveyor transfer time and shielding door opening and closing time of the radioactivity measurement system are almost constant, therefore the daily waste throughput is affected by the weight due to the waste density and the processing time according to the contamination/non-contamination determination result. In conclusion, if the weight per 200 L drum is calculated according to the density of the waste and then the waste throughput is calculated, the density of the waste is 1.5 g/cm^3 or more, the daily waste throughput is satisfied with 15 ton/day.

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Keywords: Daily waste throughput, Radioactivity measurement system, Shape measurement equipment, PVT screening equipment, HPGe scanning equipment

A Case of Clearance for Surface Contaminated Waste in Radioactive Material Use Facility

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For clearance level waste classified under domestic radioactive waste classification standard, especially in the case of volumetric radioactive waste such as object or material, about 1 kg per 200 kg of the population can be representatively sampled for radioactivity concentration analysis according to the regulatory guidelines. In addition, in case of surface contaminated waste, which is flat and smooth, mutual uncertainly between the surfaced radiation dose rate and surface contamination degree should be complementary. The radioactivity concentration can be an objective data to prove that the clearance criteria are satisfied, but the criteria of surface contaminated waste are still insufficient in Korea. Criteria for determination of surface contamination should be required if a contamination measurement is to be performed, as well as the procedures related to clearance of item, buildings and site.

The current study describes a clearance case of a device made of iron generated in a radioactive material use facility.

Mass to surface ratio of 113 pieces of waste was estimated based on the size and weight pursuant to ANSI/HPS N13.12-2013. The ratio could refer to the mean value suggested by the reference guide; however, a more conservative approach was taken by directly measuring and applying it. The radioactivity concentration was derived by applying each ratio to the gross alpha/beta actual measurement.

As a result, it could be confirmed that the maximum gross alpha/beta radioactivity concentration of the materials was 3.32 E-02 Bq/g. Considering the surfaced radiation dose rate and the measurement results of fixed/removable surface contamination for all waste samples, it was confirmed there was no contamination by radiation sources.

As nuclear related facilities will continue to generate wastes subject to clearance level wastes, it is necessary to recognize the need for comprehensive clearance guide for a variety of device and structure with surface or volume radioactive substances.

Keywords: Surface contaminated waste, Radioactive waste, Clearance, Radiological evaluation

Evaluation of Radiation Dose Exposure to Driver When Transporting Radioactive Materials Using RESRAD

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When transporting radioactive material, it is very important to evaluation the radiation dose of the radiation worker. Especially the driver is one of the very important occasional worker management targets. This is because the driver to frequent access may have the highest dose of radiation due to radioactive material.

This study evaluated the radiation dose of drivers through the commonly used RESRAD program. The exposure radiation dose evaluation was performed in 2018. It was carried out as a transportation route from each nuclear power plant to the Korea Atomic Energy Research Institute and both transportation and return routes were considered. In the case of driver, only external exposure scenarios were considered. Because driver has not considered there was no factors for internal exposure.

Each of the result of evaluating the exposure dose of the driver, the result was a maximum of 0.3 μSv/hr and a minimum of 0.015 μSv/hr. The results of dose evaluation for each power plant were 0.09 μSv/hr at the Hanbit nuclear power plant, 0.015 μSv/hr at the Hanul nuclear power plant, and 0.3 μSv/hr at the Kori nuclear power plant. This result was similar to the measured by the personal autometic dosimeter between transports.

This study is based to consider as RESRAD-Recycle or RADTRAN has not regularly service updated. In this aspect, certainly this is worth. When dismantling related facilities such as radiation control areas, it is expected that a large amount of waste will be generated and moved. In future, if the evaluation of the exposure radiation dose of driver will be carried out through various cases through RESRAD-Onsite, the reliability is expected to increase furthermore.

Keywords: Radiation Dose, RESRAD-Onsite, Driver, Radioactive Material

Identification Gamma Radionuclide Using Light-output Ratio of Inorganic and Organic Scintillators

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The scintillator-based plastic optical fiber sensor is free of electromagnetic interference, flexible and used for long-distance measurement. However, when plastic optical fiber is used for measuring gamma-ray spectra at long-distance, the full width at half maximum of full energy peak increases due to its modal dispersion. Due to poor energy resolution, it is difficult to apply conventional radionuclide identification method which finds full energy peaks using a gamma-ray spectroscopy. Therefore, we proposed an alternative method to identify gamma radionuclides at long-distance using a light-output ratio of inorganic and organic scintillators-based plastic optical fiber sensor.

In general, the amount of deposited energy in a scintillator depends on the energy and intensity of incident gamma-ray and characteristics of its density and effective atomic number. Also, the number of emitted photons is proportional to deposited energy in a scintillator. By measuring the number of emitted photons, we can estimate the amount of deposited energy indirectly. However, the measured light-output is variable depends on the type of scintillator, and we measure the light-output ratio of inorganic and organic scintillators.

In this study, we fabricated a plastic optical fiber sensor with four different kinds of scintillators. Based on the characteristics of scintillator such as light-yield, density and peak emission wavelength, we selected two inorganic scintillators such as GAGG:Ce, YSO:Ce (Epic-crystal) and two organic scintillators such as BCF-12, BCF-20 (Saint-Gobain). Each selected scintillator had a cylindrical shape with height of 15 mm and diameter of 3 mm and four of them were unified. And a 0.5 m-long plastic optical fiber (CK-80, Mitsubishi Rayon Co., Ltd.) with a diameter of 2 mm was attached to the bottom part of each scintillator. The total size of the sensor-tip was a thick of 1.9 cm, a width of 1.9 cm and a height of 2.6 cm. As a light-measuring detector four photon counting modules (H11890-210, Hamamatsu photonics) were used simultaneously. The ^{241}Am, ^{137}Cs and ^{60}Co gamma-ray irradiator were used as a gamma-ray source and their emitting gamma-ray energies were 0.0595 MeV, 0.662 MeV and 1.173, 1.332 MeV, respectively. The gamma-ray intensity was adjusted by the distance using a rail attached to the irradiator.

We found that each scintillator emitted photons that was proportional to the intensity of incident gamma-ray and it followed an inverse square law exactly. The light-output ratios of four kinds of scintillators were measured and they were reversely proportional to the energies of incident-gamma rays. The difference of the light-output ratio is more dominant in the combination of inorganic and organic scintillators. From these results, three gamma-emitting radionuclides could be identified.

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Keywords: Light-output ratio, Radionuclide identification, Scintillator, Energy dependency

A Study on the Design of a Mobile Compression Facility for Decommissioning Wastes of Nuclear Power Plants That Can be Used for Dual Purposes

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The dismantling of nuclear power plants is increasing worldwide due to the aging of nuclear power plants, and the dismantling of nuclear power plants is expected to proceed gradually in Korea starting with Kori No. 1. Many of the wastes generated when dismantling nuclear power plant structures are medium- and low-level wastes, which are mainly processed through compression. In general, a method of directly compressing waste using a low pressure compression facility and a method of directly compressing a waste drum using an ultra-high compression facility is used for waste compression. Currently, ultra-high compression equipment can only be used as a fixed type, so users may be exposed to radiation, and it is difficult to directly compress waste. Low pressure compression equipment is difficult to use when high pressure is required, such as waste drums compression.

In this study, the design of a movable compaction facility that improved the shortcomings of the existing compression facilities was conducted. The super-compaction module aims to be installed in a 20 ft ISO container. The reduction module consists of a conveyor assembly that transports waste to a container, a press assembly that compresses waste, and a horizontal conveyor assembly that transports compressed waste. The press assembly was designed to allow both direct compression of waste and recompression of waste drums according to the purpose of the compression facility. The compaction facility designed in this study is considered to be more economical and safer than the existing compression facilities in the future treatment of nuclear power dismantling waste.

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Keywords: Decommissioning waste, Waste volume reduction, Movable waste reduction system, Super-compaction

Operating Characteristics of Remote Lid Fastening Device for High Integrity Container

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A remote lid fastening device of the HIC (High Integrity Container) is under development for disposal of the high dose spent resins from the nuclear power plants. The surface dose of the HIC loaded with the high dose spent resins stored by the nuclear power plants is expected to be excessive exposure for the operator to package, transfer, and dispose of. Therefore, it is necessary to develop a remote lid fastening device to reduce exposure when handling high-dose spent resins in the HIC. The remote lid fastening device consists of a drive motor, a bundle of gears, a rotating plate of the lid, and high integrity container tongs. For monitoring and measurement, two image monitors are installed, with a radiation monitor that can measure 0.1 to 100 mSv/hr, enabling confirmation of fastening and surface dose measurement before and after lid fastening. The remote lid fastening device allows automatic operation, manual operation, and proper isolation of the operation control panel to reduce the radiation exposure of workers. The installation and removal of the automatic lid fastening device of the high integrity container shall be carried out by the operator using the existing crane, and the lid shall not exceed 30 minutes for a single fastening. All works of lid automatic fastening device were done remotely to solve the problem of worker exposure and designed to be automatically moved to the correct fastening position by monitoring cameras and high integrity container tongs. Currently, the design and technical specifications of the remote lid automatic fastening system are derived, and detailed design is being carried out based on the relevant technical specifications by deriving general design requirements, quality requirements, and measurement/control requirements. This technology development will contribute the data requirements of the HIC operating licensing for high-dose spent resins and the disposal of the high integrity containers.

Keywords: Radioactive waste, Remote lid fastening device, HIC (High Integrity Container), High dose spent resin

A Hybrid Modeling Approach to Evaluate Exposure Dose Rate at the LILW Disposal Site in Korea

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Radioactive wastes generated during operational phase and decommissioning of a nuclear facility should be managed in safe ways in order to protect human health and minimize the environmental impact. Radionuclides contained in the radioactive wastes will be eventually released and transported to the accessible environment (near-field, far-field, and biosphere) from disposal facility. Therefore, based on the characteristics of radioactive waste, an appropriate safety assessment modeling has to be implemented and the long-term safety assessment of radioactive waste disposal should be described by potential releases of radionuclides from the disposal site. In this study, we suggest appropriate hybrid modeling methods for the safety assessment of the low- and intermediate-level waste (LILW) disposal facility as three steps: 1) Input parameters (effective diffusion coefficient, diffusivity) for the modeling are mainly obtained from leaching experiments using solidified radioactive wastes, 2) COMSOL Multiphysics with radionuclide transport module are used to simulate the release and transport of radionuclide and calculate average discharge rates for radionuclide on the near-field, and 3) Safety assessment and performance evaluation of exposure dose rate on the far-field are developed by utilizing GoldSim with a radionuclide transport module.

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Keywords: Radioactive waste, Safety assessment, Low- and intermediate-level waste disposal facility, COMSOL Multiphysics, GoldSim

A Disposal System of Liquid Waste Generated in Separate Buildings of NPPs

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The liquid radioactive waste system in NPPs is designed to collect, store, process and dispose of radioactive or potentially radioactive waste for recycling to the reactor coolant system or for releasing to the environment. These wastes are either recycled for reuse, released under controlled conditions via the service water system, retained for further processing, or transferred to the solid waste system for solidification. However, liquid waste generated in a separate building should transfer to the main building for treatment. It is not only inefficient that liquid waste must be transferred to the main building every time when measurement value higher than the standard value. It is not desirable to install and operate an expensive radioactive waste treatment facility in a separate building. In order to solve this problem, it is intended to manufacture and use a liquid waste treatment facility using an ion exchange resin in a separate building.

Component of Apparatus

- Pump (PW-S354SMA) for collecting liquid waste
- A tank for stabilizing of radioisotope by adjusting the PH and storing liquid waste temporally collected through a pump
- The first activated carbon filter that absorbs and filters radioactive isotopes in liquid waste that moves after being stored in the tank
- Cation exchange resin that exchanges and removes cations in waste moving through the first activated carbon filter
- Anion exchange resin that exchanges and removes anions in waste moving through the first activated carbon filter.
- A second activated carbon filter re-adsorbing and filtering radioactive isotopes.

Although there may be a difference in the decontamination coefficient depending on the initial concentration, the decontamination rate was approximately 10^3. A method to efficiently treat liquid waste generated in a separate building within a nuclear power plant was considered, and a liquid waste treatment apparatus using the ion exchange resin as described above was developed. It is believed that the use of the above-described treatment facility instead of the conventional way will greatly reduce the frequency of moving liquid waste from a separate building to a nuclear reactor building. It is considered that the management system can be used as a reference when establishing the standards for liquid waste treatment for separate buildings.

Keywords: Liquid waste, Ion exchange resin

The Perspectives of the Radioactive Waste Data for Data Science and Predictive Modeling

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Many fields in industry have been applying data science and artificial intelligence (AI) for explanation and prediction of phenomena using the big data which has a couple of well-defined characteristics. Likewise, the data, presumedly big, from decommissioning and operation of NPP is generated during treatment/disposal of radioactive waste, which should be managed and maintained for a long-term, from generation to generation, through Waste Tracking System accordingly. Utilizing this big data being measured, accumulated and qualified continuously, it is worth developing the statistical machine learning and data science models such as prediction, classification, and clustering, and so forth. This paper outlooks it from the perspectives of data science and partly conventional statistics, especially with respect to the "data preparation and wrangling" to support the model analysis and model building along with machine learning life cycle.

The development of AI models following such data investigation and preparation will be of great help in deploying the waste management and policy based on the infrastructure tentatively called as "*the integrated standard bigdata platform*".

Keywords: Data science, Bigdata, Long-term management, Data wrangling, Waste tracking system (WTS), Machine learning, Standard bigdata platform

A Study on Induction Melting for Metallic Radioactive Waste in Decommissioning of NPPs

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A large amount of the radioactive wastes is generated during decommissioning of Nuclear Power Plants (NPPs). Because the metallic radioactive waste which is mostly very low and low level is expected to be about 60 percent of the radioactive waste, it is important to reduce the amount of the metallic radioactive waste. The induction melting (IM) system is one of efficient method for radioactive waste treatment and decontamination. Moreover, the device is simple to use, and the ingot is homogenized by convection. Hence, we manufactured the IM systems for volume reduction (20 kg) and the analysis of the nuclide transport (2 kg), respectively. The IM systems consisted of a 100kW Power supply, a 20HP Cooling system, two-capacity melting furnace, exhaust treatment equipment, control panel. In addition, the exhaust treatment equipment was composed of two types (Pre & Hepa) of heat-resistant filters and 285 mmAq Blower in preparation for hot fume generation. To estimate the volume reduction ratio by comparing the volume, the carbon steel (SS40), the stainless steel (SUS 304), and the inconel (Alloy 600), which were most used in the primary system were used. The stable nuclides such as the Co and Cs were used to simulate the nuclides of gamma key. Finally, the homogenization of ingots and analysis of the nuclide migration in ingot, slag, and filters were evaluated using X-Ray Fluorescence (XRF) system.

Keywords: Decommissioning, Homogenization, Volume reduction, Induction melting, Radioactive waste treatment

A Study on Gas Generation of Radioactive Wastes in High Integrity Container

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This study describes the evaluation of the gas generation of radioactive waste in High integrity container (HIC). The disposal characteristic environment can be largely assumed in two ways to storage environment and operating environment. During the operation of the disposal facility, the environment around the waste storage area must be kept dry with constant temperature and humidity. In addition, since oxygen is diffused through gaps in the package or through the exhaust system, almost all the package maintains an aerobic environment during the operation period of disposal facility. And it is assumed that the anaerobic environment is maintained with cement filler. Under the constant temperature and humidity conditions of each building, the inside of the waste is maintained in aerobic and dry conditions, so it is expected that there will be little gas generation in the waste during the operation period. Level 2 near surface disposal facility will significantly reduce gas permeability since cement grouting is carried out after loading the first stage in the disposal facility. However, the reduction effect was not conservatively considered in the evaluation.

The mechanisms of gas generation in waste are corrosion of metals, microbial decomposition of organic matter, and radiation decomposition of water and organic matter. The waste in the HIC was assumed to be spent resin and concentrated waste. In spent resin, gases such as ^{3}H and ^{14}C ($^{14}CH_4$, $^{14}CO_2$) are generated by metal corrosion, radiolysis, decomposition of organic matter, and diffusion mechanisms in the metal. In the case of concentrated waste, gases such as ^{3}H and ^{14}C ($^{14}CH_4$, $^{14}CO_2$) are generated by metal corrosion, radiolysis, and diffusion mechanisms in the metal. The amount of radioactive gas emitted by waste decomposition is divide into light water reactor and heavy water reactor waste. And it was assumed conservatively to be discharged as tritium (^{3}H), carbon dioxide ($^{14}CO_2$), and methane ($^{14}CH_4$) by dividing it into drums and HICs.

In this study, a methodology for evaluating the amount of gas generated was proposed in consideration of the disposal characteristics environment and the gas generation characteristics of waste. Through this, it is planned to evaluate the gas generation level inside the HIC and review the necessity of the exhaust valve.

Keywords: HIC, Gas generation, Radioactive waste

Suggestion on Selection of AI and Optimization Techniques for Deriving the Optimum Emplacement Scheme of the LILW Complex Disposal Facility

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For disposing of low- and intermediate-level radioactive waste (LILW) in our country, the disposal center with a total capacity of 800,000 drums (on the basis of 200 L) will be phased in several stages in accordance with the comprehensive development plan. Since the radioactive waste has different properties for each wasteform and package, its effect is inevitably dependent on the facility features and/or location of the waste package being disposed of. In the aspect of the operational and post-closure safety, it is obvious that each waste package should be emplaced in the optimum location where the objective function (e.g. minimization of annual exposure dose to the critical group) defined by the radioactive waste management agency can be optimized in consideration of characteristics such as geometrical configuration, radionuclidic inventory, release and transport mechanisms, and etc. Whether the performance goal of the disposal facility is fulfilled can be judged by performing the safety assessment, and ideally provided that characteristics of all the waste packages being disposed of are definitely identified prior to the operational phase, it is possible to derive the optimum emplacement scheme for the entire facility through analyzing these results.

However, in practice, there are some limitations as follows. First, multiple computer codes need to be used for deriving value of the objective function to be optimized for each emplacement, and it takes a lot of time to calculate the value. Second, it is impossible to compare all emplacement options with each other due to the vast number of cases. Third, not all the drums are disposed of simultaneously.

To solve these challenges, in this study, applicability of artificial intelligence (AI) and optimization methodologies to formulate the emplacement optimization strategy were investigated, machine learning algorithms (such as convolutional neural network (CNN), recurrent neural network (RNN), and reinforcement learning) for the former, and function optimization, combination optimization, and heuristic algorithm for the latter, respectively.

As a result, it was considered that the simulated annealing (SA) algorithm was the most suitable for the optimization methodology for deriving the optimum emplacement scheme of the complex disposal facility. By applying SA, the approximate optimum value can be found without comparing all the possible cases. Even though there are other considerations, results of this study are expected to be utilized for implementing the emplacement optimization of radioactive waste in the complex disposal facility in conjunction with the disposal safety assessment program using GoldSim.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (20193210100110, Development of Construction and Management Optimization Technique for the Complex Disposal Facility).

Keywords: Complex disposal facility, Emplacement, Optimization, AI, Simulated annealing

Review on Thermal Treatment Technologies for Liquid Wastes Generated From Multi Nuclide Removal Facility

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Advanced Liquid Processing System (ALPS), a multi nuclide removal facility, is in operation to treat contaminated water from the Fukushima nuclear power plants accident. Advanced Liquid Processing System consists of a pretreatment process and an adsorbent process. The pretreatment process consists of iron co-precipitation and carbonate precipitation. Iron (III) hydroxide process has the function of removing Co-60 and Mn-54, including alpha nuclides. In the carbonate precipitation process, Mg and Ca, which are ions that inhibit adsorption performance, are removed, and Sr-89, Sr-90, etc. are precipitated. In the adsorbent process after the pretreatment process, activated carbon is the first, where colloidal nuclides I-129 and Co-60 are adsorbed. Second, Sr adsorbent adsorbs Sr-89 and Sr-90. Third, Cs adsorbent adsorbs C-134 and Cs-137. I/Sb adsorbent adsorbs I-129 (IO^{3-}) and Sb-125. Fourth, in I adsorbent, I-129 (I^{-}) is adsorbed. Finally, Ru adsorbent adsorbs Ru-106. The pretreatment and adsorbent wastes generated from ALPS are stored in HIC (High Integrity Container) in the form of slurry and/or sludge including water. High temperature thermal treatment technologies are being considered as suitable methods for the final disposal of Advanced Liquid Processing System slurry and sludge wastes. Among them, a feasibility study on the applicability of vitrification technology is ongoing as the most promising treatment technology.

Keywords: ALPS, Pretreatment, Co-precipitation, Adsorbent, Vitrification

Analysis of the Elements to Build an Archive Hub System for Radiochemical Analysis of Radwaste Samples

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The amounts of low and medium-level radioactive wastes to be generated during the decommissioning of Kori Unit 1 are estimated about 14,500 drums. They are mainly contaminated soils, concrete debris and metal scraps, which their matrices are very complex, the radioactivity range being wide. Therefore it will be difficult to distinguish the deregulated wastes from the regulated wastes if they are not managed systematically through the process of sample receipt, inspection, tracking and radiochemical analysis. Besides, in the course of receiving and inspecting these samples with various radiochemical properties and analyzing the radioactivity of 14 radionuclides including ^{3}H and ^{14}C, the risk of having to re-sampling and re-analysis can be significantly increased due to the following mistakes: an inability to trace the sample by writing an incorrect sample number, a shortage of samples by practitioner's fault and an occurrence of outliers by using inhomogeneous samples, etc.

This study focused on a system which can track and manage a sample history in the process from receiving on-site samples to issuing final test reports. In addition, after dividing the samples into a Working Half for a short-term storage and an Archive Half for a long-term storage, we investigated the basic elements and cases needed to build an Archive Hub System (AHS) that can safely store this Archive Half in a stable Atmosphere. Finally, KS Q ISO/IEC 17025 standard was analyzed, and then several requirements were extracted for improving the reliability of the Archive Hub System.

Keywords: Archive Hub System, Radiochemical analysis, Radwaste samples, Decommissioning of Kori Unit 1

Design of Plasma Arc Melting-Molten Salt Oxidation Process for the Treatment of Hazardous Radioactive Wastes

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In terms of environmental and final disposal safety, radioactive waste which containing harmful substances must be treated in an appropriate way to remove its hazardous properties. In this study, we developed an eco-friendly process that has the advantages of the existing high-temperature heat treatment process and is optimized for the treatment of hazardous radioactive wastes. This process is a Plasma Arc Melting-Molten Salt Oxidation (PAM-MSO) hybrid system. The melting device destroys and melts both organic and inorganic substances at over 1500℃ in a very stable condition. The melting furnace is equipped with three graphite electrodes on the top and generates an arc between the waste and each electrode. The waste around the electrode starts to melt by initial arc energy. The energy is continuously supplied to the waste by arc generation or Joule heating by adjusting the height of electrodes up and down. The halogenated substances, heavy metals, volatile radionuclides, organic gases in the off-gas are subsequently oxidized and chemically react with molten salts inside the MSO which operates at around 800℃. The overall off-gas treatment system is composed of cyclone, MSO, cooler, AC/HEPA filter. The PAM-MSO process takes the advantage of being able to apply to various types of hazardous wastes such as combustible, non-combustible, and liquid and to finally produce a safe and stable waste form. This simpler and compact process is assumed to greatly reduce the construction and operation/maintenance cost. In this study, a bench-scale PAM-MSO system was developed and designed to prove its performance of effectively treating hazardous radioactive wastes.

Keywords: Hazardous Radioactive Waste, Plasma Arc Melting, Molten Salt Oxidization, Waste Treatment

A Preliminary Development of Arc Melting Off-gas Treatment System With Molten Salt Oxidation

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This paper presents to demonstrate the functionality and viability of the newly developed arc melting off-gas treatment system. The system main components consist of the Electrode Arc Melting furnace, Cyclone, Molten Salt Oxidation (MSO), High Efficiency Mist Eliminators (HEME) and High Efficiency Particulate Airfilters (HEPA). These components are generally applied to the business industry for the off-gas treatment process except the MSO. The MSO is designed to treat wastes by using chemical and thermal methods, which decomposes harmful metals through flameless oxidation reaction and simultaneously keeps inorganic and heavy metals in a molten salt. This characteristic prevents the discharge of acid gas, radioactive waste and hazardous heavy metals. The results of the residual amount of harmful gas from sampling ports of each component have been indicated in this paper. The harmful gas emission was expected to be below the regulation limits. We anticipated that the MSO effectively eliminates harmful gas on the current off-gas treatment system.

Keywords: Molten Salt Oxidation (MSO), Arc melting off-gas treatment system, Flameless oxidation reaction

Mechanical Safety of Very-Low-Level-Waste Transport & Storage Containers

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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. ABAQUS/Explicit, a commercial finite element analysis program, was used for mechanical safety analysis of transport and storage container.

In the case of a VLLW transport and storage container, the structural integrity of the container, such as shielding and stability, should be verified by carrying out a drop test and a stacking test under normal transport conditions. Among the normal transportation conditions, a 1.2 m high free fall shock analysis, which is a hypothetical accident condition that may damage the container, and a stacking analysis for a load 5 times the amount of transported material should be performed.

As result of the drop test explanation, the maximum stress when colliding with the ground in the drop condition in the 45˚ inclination direction of the container was calculated as about 794 MPa, and the maximum stress point was found at the edge axis of the upper cover. Although plastic deformation occurred at the corresponding edge axis, it was evaluated that the range of plastic deformation was limited to the upper cover and cage, and the inner container had stress within the elastic limit. In the analysis results of other fall direction conditions, the stress due to collision with the ground and the load of the container affects each structure, but it was evaluated that the inner container generated stress within the elastic limit except for minor deformation.

In the lamination analysis, when analyzing the lamination condition for a load 5 times the weight of the container, the axial stress of the upper cover, cage and pallet subjected to each load did not undergo plastic deformation. The maximum displacement for the loading load was 0.0432 mm, and it was evaluated that a very small displacement occurred.

Therefore, it was confirmed the shielding and structural integrity of the container through drop and lamination analysis of the container according to the technical standard of the container.

Acknowledgements

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Keywords: VLLW, Liquid radioactive waste, Transport and storage container, Safety analysis

Derivation of Pyrolysis Operating Conditions for Volume Reduction of Radiation Waste

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The ubiquitous technologies such as compression, incineration and pyrolysis etc. are used to reduce the volume of radioactive waste. This paper is to describe the test results performed to derive the operating conditions for thermal decomposition for each sample waste, which could be experimental data to reduce the volume through pyrolysis.

Most of the radioactive waste generated during nuclear power plant operation in places are dry active waste such as clothing and gloves used during work. Therefore, clothing, gloves(latex), paper, wood, PVC (Polyvinyl chloride), and PE (Polyethylene) were selected as experimental samples to simulate and test thermal decomposition of those wastes.

The test was performed by changing the operating temperature and the maximum temperature holding time using the Lab Scale equipment, deriving the operating conditions for each sample waste, and measuring the ratio of weight change for each experiment. In case of PE (Polyethylene) that was evaporated to generate a large amount of gas, so it was impossible to reduce the volume through pyrolysis. For the remaining samples other than PE, the ratio of weight change was measured to be about -73.83% to -84.80%.

Keywords: Thermal decomposition, Pyrolysis, Volume reduction, Operating conditions, Operating temperature, Maximum temperature holding time

Development of 100 kW Plasma Torch Melting System for Radioactive Waste

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100kW class plasma torch melting system was designed for developing the treatment, volume reduction and disposal technology of radioactive wastes. As a pilot test facility for treating of various radioactive waste including the combustible, un-combustible, inorganic and liquid type wastes, it mainly consists of melting chamber, pyrolysis chamber, slag discharge, waste feeding system, and maintenance man-hole. In addition, the off-gas treatment system will be installed for law of Air Quality Conservation Act Preservation by removing the NO_x, SO_x, CO_x and etc. of off-gas. The off-gas system is primarily composed of 2nd combustion chamber, SNCR (Selective Non-catalytic Reduction), centrifugal-bubbling, and ID (Induced-Draft) fan.

As of peculiar feature of 100 kW plasma torch melting system, overflow discharge method according to the level of molten is being considered. For the discharge method, it is expected to be easily maintenance and long term integrity of the plasma torch melter as its advantages. Also, varied waste feeding devices such as pusher, spray nozzle were designed in order to treat the various waste types.

Currently, the 100 kW class plasma torch melting system is being constructed in KHNP CRI. KHNP CRI plans to evaluate the applicability using the simulated wastes.

Keywords: 100 kW Plasma torch melting system, Radioactive waste, Overflow discharge method, Varied waste feeding devices

Development of Discharging Nozzle for Cold Crucible Induction Melter

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Vitrification with cold crucible induction melter (CCIM) is considered as one of promising treatment technology for radioactive wastes such as combustible and liquid type wastes. Vitrification technology is to chemically or physically incorporate radioactive nuclides in the glass matrix as radioactive wastes are fed on the molten glass. Accordingly, release of radioactive nuclides to the environment at the disposal site can be fundamentally prevented. In the Central Research Institute of Korea Hydro and Nuclear Power, vitrification demonstration test has been performed to evaluate the applicability for secondary radioactive wastes generated by contaminated water treatment facility of Fukushima Daiichi Power Station.

In this study, developed discharging nozzle among the CCIM components was evaluated in view of the integrity of discharging nozzle on operation environment (above 1,100 degree Celsius on air environment). Four discharging nozzles are respectively made of different materials which is Carbon, Titanium alloy, stainless steel 310 and Inconel 690. For carbon and titanium nozzles, the integrity of structure was significantly degraded by high temperature oxidation environment. The discharging nozzles made of stainless steel 310 and Inconel 690 are being fabricated to the evaluation of their integrity.

Keywords: Vitrification, Cold crucible induction melter, Discharging nozzle, Materials

Evaluation of Applicability for Metal Waste Using MW Class Plasma Torch Melting System

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Plasma melting has been considered as promising technology for volume reduction and treatment of radioactive wastes. As advantages of this technology, IAEA TECDOC-1527(2006) report that the plasma torch melting technology can treat regardless of waste types which is both metallic and non-metallic wastes. Furthermore, plasma torch melting technology is expected to produce a great volume reduction ratio compared that of with other technologies such as super compaction. In order to put the advantages of plasma torch melting technology to commercial use, many countries such as Switzerland, Bulgaria, Japan and China have supported the development and construction of plasma torch melting facility for the treatment and disposal of radioactive wastes generated from operation and decommissioning nuclear power plant.

A MW (Mega Watt) class plasma torch melting facility was designed and constructed at Central Research Institute of Korea Hydro and Nuclear Power as a pilot test facility to evaluate the applicability for treatment of radioactive wastes. This facility has the throughput of maximum 250 kg per hour for the metals. In this study, plasma melting test was performed to evaluate the applicability for such as stainless steel and carbon steel among of waste types. Also, the integrity of plasma torch by the exposure at high temperature in air environment was evaluated.

Keywords: MW class plasma torch melting system, Volume reduction, Radioactive wastes, Plasma melting

Review on Glass Melt Discharge Technologies Applied to Vitrification Technology

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When liquid and/or solid radioactive wastes are vitrified at 1,000ºC or higher and poured into a metal canister to solidify, the core process is to discharge glass melt continuously or in batch type. Among the glass melt discharge devices, there is a case of discharging by tilting the first, but it is not applied very much. Second, there is a method of discharging by opening the outlet using a physical tool. In this case, it will be a method of operating by installing a sliding gate. Third, there is a method of discharging by heating the area around the discharge port, and this method is one of the most widely used methods in commercial vitrification facilities at present worldwide. Among the methods of directly heating and discharging the area around the outlet, a resistance heating coil is installed to heat the surrounding temperature. The induction heating method is a method in which the temperature is raised by directly heating the discharge metal pipe, and when the temperature is raised, the glass inside the discharge metal pipe is melted and discharged. In the meantime, experiences of applying various discharge methods applied in vitrification technologies using ceramic lined glass melter and CCIM (Cold Crucible Induction Melter) were shared, and the advantages and disadvantages of each technology were evaluated.

Keywords: Glass melt, Discharge, Sliding gate, Induction heating

Case Study for Establishing Strategy of Predisposal Management of Radioactive Waste Using System Dynamics

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Predisposal management of radioactive waste covers all the steps in the management of radioactive waste from its generation up to disposal, including processing, storage and transport. Optimized strategy of predisposal management can be derived by performing safety assessment, cost analysis, qualitative analysis and so on at each step. Safety assessment and cost analysis of predisposal management can be respectively undertaken using “SADRWMS (Safety Assessment Driving Radioactive Waste Management Solutions)” methodology developed by IAEA and “ISDC (International Structure for Decommissioning Costing)” methodology developed by OECD/NEA. However, since existing methodologies of analysis are evaluated by assuming a specific scenario, flexible evaluation is impossible and if assumption conditions change, all of analysis should be newly undertaken. System dynamics is composed of variables related to a problem and dynamically analyzes the relationship between each variable with computer modeling. Dynamic analysis method can handle flexibly to changing situations and overcome the limitations of existing methodologies. Therefore, in order to derive an optimized predisposal management through methodology of system dynamics with existing methodology, case study of system dynamics methodology for optimal management of radioactive waste is investigated. System dynamics was used by the LLWR (Low Level Waste Repository Ltd) to better understand and quantify the national LLW system and to support detailed analysis of a particular type of LLW. System dynamics model consists of three stages, the first stage of a System dynamics based project involves mapping the cause and effect relationships that drive system behaviour. Once an agreed diagrammatic representation of the system has been created, specialist software can be used to quantify the relationships. The quantified model can be evaluated for range of containers used for the transportation and storage/disposal of the LLW and the potential waste treatment methods and their impact on resources. KAERI is developing a simulator (“ENVI”) to allow decision-makers to rapidly configure and evaluate the performance of alternative national strategies for SNF (Spent Nuclear Fuel) management. The simulator represents, in a simplified but integrated way, all activities associated with SNF management from the time of its origination in a power plant to its final disposal. Two case studies which applied system dynamics are investigated to derive optimal predisposal management in this study.

Keywords: Predisposal Management, System Dynamics, Optimization, Radioactive Waste

Evaluation of the Decontamination Efficiency of Radioactive Wastes Generated During the Production of ^{201}Tl

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This study evaluated the rate of radioactivity removal of each decontamination substance by using four types of decontamination materials for intermediate and low level radioactive waste generated by using cyclotron at KIRAMS.

The intermediate and low level radioactive waste used in the experiment used lead plates mounted on fume hoods used for the distribution of radioactive isotopes. An easily available decontamination treatment method at the laboratory scale was adopted as a treatment method for reducing radioactivity in radioactive waste. And water, alcohol (75% Ethyl alcohol), decontamination water (Spray mist #005-400, BIODEX Medical System INC, U.S.), and decontamination gel (DECONGEL 1101 GEL, Cellular Bioengineering Corporation, U.S.) were used as materials for decontamination treatment. Based on the previous study, the front and rear surfaces of the lead plate were washed for 1 minute each, and the surface dose rate and radioactivity were measured using a detector. (In the case of decontamination gel, it was decontaminated by applying it to the entire lead plate and then removing it.)

For the evaluation of the results, in this study, an HPGe semiconductor detector (IP25-2, CANBERRA Inc, U.S.) and a radiation spectroscopy analyzer (DSA-1000, CANBERRA Inc, U.S.) that can measure the radioactivity of radionuclides contained in waste were used. In order to measure the surface dose rate emitted from the surface of the lead plate, a digital survey meter (FH-40-L/FMZ 732GM, Thermo Fisher Scientific, Germany) was used. As a result of measuring the lead plate after decontamination using detector, a lot of radioactivity was removed in the order of decontamination water–alcohol (75% Ethyl alcohol)-decontamination gel–water. In particular, when decontamination was treated using decontaminated water, radioactivity was removed to a level approximately the concentration of self-disposal regulations.

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Keywords: Decontamination, ^{201}Tl, Cyclotron

Development of Mixed-waste Melting Furnace System

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1 MW plasma torch and melting furnace demonstration facilities are currently under development for the disposal of nuclear power plant decommissioning wastes and non-compliance wastes.

Nuclear power plant decommissioning waste and non-compliance waste have various forms, including metals, concrete, soil and hazardous waste.

In order to dispose of waste using plasma torch facilities, it is necessary to develop a melting furnace system with improved input methods according to the type of waste. Mixed-waste melting furnace system is being developed with the aim of disposing of liquid, solid and drum-type waste.

Solid wastes such as metal and concrete are designed to be injected into the melting furnace through the rotation of the screw using the motor. Liquid waste, such as sludge and waste oil, is designed to be used inside the melting furnace by controlling the speed and flow rate by using a pump in the storage tank. In the case of waste drums, the pusher is used to feed the drums. The waste drum is crushed using a small capacity plasma torch before being used to melt the waste drum into the furnace in a relatively small shape.

It is expected that the development of a mixed-waste melting furnace system will secure the advancement of the disposal technology of nuclear power plant decommissioning waste in the form of solid, liquid, and drums.

Keywords: Plasma torch melting system, Decommissioning waste, Mixed-waste melting furnace system, Drum waste

Development of the Dam-type Discharge Device for Plasma Torch Melting System

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Kori unit 1 began commercial operations in 1978. It was decided to permanently shut down in 2017. As a result, interest in the disposal of decommissioning waste is increasing. KHNP's CRI is conducting research for the volume reduction and disposal of decommissioning waste by using a demonstration facility for waste disposal using a 1 MW plasma torch. It is necessary to develop the Dam-type discharge device that can discharge stable molten slag and or metal to volume reduction and dispose of decommissioning waste.

The Dam-type discharge device is a discharge device in which when waste inside the melting furnace is melted by a torch, the molten slag and or metal overflows and discharges. A typical discharge device opens a valve installed at the bottom of the melting furnace to discharge the molten slag and or metal. A disadvantage of this is that opening a valve installed on the underside of the melting furnace may cause the discharge to fail if the temperature of the molten slag and or metal is low. The dam-type discharge device is designed to flow naturally when the molten slag and or metal is filled inside the melting furnace, improving the discharge function from the existing valve-type discharge system. In addition, it is designed to have fewer failure elements because it is in a overflowing method rather than discharging by valve operation.

The development of the dam-type discharge device will ensure the safety of the plasma torch melting system.

Keywords: Plasma torch, Decommissioning waste, Dam-type discharge system, Molten slag and or metal

Development of Multipurpose Plasma Melting System

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This facility was developed to secure the melting characteristics and the operational technology for target objects composed of non-combustible materials such as concrete, ore, metal, or others. It aims to characterize the melting properties of non-combustible materials, recover metals after melting and concentrating, anddevelopcritical technology to synthesize nano-sized materials.

The facility consists of several subsystems: a transferred thermal plasma torch system as a heat source, a melting system, a gas cleaning system, and an automatic control system. The maximum power of the torch is 100 kW, and the specification of the furnace is determined according to it.The melting system is designed to discharge the molten pool continuously by applying the overflow method.As a result, molten layers can be recovered selectively depending on the operation conditions. Besides, the emission of pollutants is minimized by burning the exhaust gases in the next combustion chamber and by applying an SNCR, a scrubber, and etc.

In this paper, ores as melting objects are prepared: they are a mixture of oxide compounds such as CaO, MgO, SiO_2, and Al_2O_3 and metals such as Fe, Pd, Rh, Pt, and others.

Keywords: Plasma System, Torch, Concrete, Melting

Development of the Rod-type Discharge System for Vitrification Facility

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The recent earthquake in Japan has raised interest in the disposal of radioactive waste after the Fukushima nuclear power plant accident. At KHNP CRI, research is underway on the disposal of radioactive waste using vitrification technology. It is necessary to develop a discharge device using the rod that can be discharged periodically for stable operation of vitrification facility.

The Rod-type ejector physically blocks the outlet inside the vitrification melting furnace. It is designed to withdraw the rod when the discharge of glass melt. As a detailed design, the rod in the melting furnace was designed in a water-cooled method to prevent damage. The exposed part of the rod was alumina coated to prevent the rod from being induced by electromagnetic fields. The Rod-type discharge system is designed to be fixed to the top of the melting furnace of the vitrification facility and secured without shaking. The operation of the rod is performed using air pressure to perform withdrawal and insertion. Normally, the rod is inserted to block the furnace outlet. It is designed to discharge glass melt through the withdrawal of the rod.

It is expected that the development of the rod-type discharge system will increase the efficiency of vitrification facility operation through periodic discharge.

Keywords: Vitrification, Radioactive waste, Rod-type discharge system, Glass melt

Evaluation of Equivalent Drop Height for Radioactive Wasting Packaging Containers Using DIC Method

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In this study, the drop height of the radioactive waste packaging container was obtained by applying the DIC method to perform vertical drop test to produce structural integrity and analysis verification data under general conditions of the radioactive waste packaging container. radioactive waste packaging containers are not prescribed in the Nuclear Safety and Security Commission's notice, so arbitrary normal conditions are assumed. The drop height (30 mm) is determined by considering a safety factor in the maximum operating speed of the crane. In particular, all the materials in the container have to be within the elastic range. But Because of the friction at the release mechanism for picking up and holding the container, the container is collision to the rigid target at a lower speed than the calculated speed (velocity $= \sqrt{2gh}$). Therefore, in this study, we derive equivalent drop height such as crane operating speed through repeated tests using Digital Image Correlation (DIC) techniques.

Finally, a structural integrity assessment was performed for the condition under which the container was dropped at a drop height of 50 mm, and the results confirmed its safety.

Keywords: Drop height, Package container, Integrity

Development of Sampling Device for Radwaste Characterization Specimen

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In this study, a sampling device for radwaste characterization specimen was developed. 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.

Although there are many ways to process the specimen for characterization, for taking representative specimen with some quality in a radiation field, a device should be developed. For development of a device, we considered some factors such as safety accidents due to heavy drum handling, limitation of water use for minimization of secondary waste generation. Additionally, maintaining of specimen quality such as constant dimension and its integrity was also considered. To solve this problem, the main body part, core bit part, dust collector, compressed air, and control part were constructed to automatically collect specimens from the drum by remote control by dry method. This development has also been registered as a domestic patent and is expected to be used in the field of radioactive waste disposal industry in the future.

Keywords: Solidification, Radioactive waste, Drum, Core drill, Characterization

Review on International Radwaste Acceptance Criteria for Thermal Treatment Technology

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Various technologies are being developed for the stable treatment of radioactive waste in many countries. Among them, the thermal treatment technology of radioactive waste is being intensively studied in many countries because it can treat a variety waste. Radioactive waste treated through thermal treatment technology should be evaluated for disposal suitability for final disposal. And radioactive waste disposal sites need to establish waste acceptance criteria for acceptance of such conditioned radioactive waste.

The KORAD established acceptance criteria for wastes treated with vitrification and reflected them in waste acceptance criteria (WAC-SIL-2020-1). In KHNP, there is a possibility to apply not only vitrification but also other thermal treatment technologies, so it is necessary to analyze the waste acceptance criteria. Unlike the United States, in Europe, which has a disposal environment similar to that of Korea, there are established waste acceptance criteria for each country. It is necessary to analyze trends in the acceptance criteria of thermally treated wastes in each country's radioactive waste disposal site. Representatively, the review of waste acceptance criteria of NIRAS/ONDRAF of Belgium and KONRAD of Germany is conducted.

Keywords: Radwaste, Waste acceptance criteria, Disposal, Thermal treatment technology

Development of Integrated Decontamination Process With Remote Control Technology Applied

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In the case of Korea in a situation where it is difficult to predict the amount of decommissioning waste due to no experience in decommissioning commercial NPPs, Gori Unit 1 was decided to decommission. The decommissioning cost is estimated at 812.9 billion won, of which 254.1 billion won as radioactive waste management expenses. According to the Nuclear Power Plant Decommissioning Source Term Evaluation Technology Development Report (KEPCO E&C, 2016), if Kori Unit 1 is immediately dismantled, 34,200 drums of radioactive concrete waste are expected to be generated. The low- and intermediate-level radioactive waste management expenses is 15.19 million won (based on 200 L drums). And the management expenses of radioactive concrete waste generated during dismantling Kori Unit 1 is expected to be 519.5 billion won. This can lead to delay in decommissioning and thus radioactive concrete waste reduction technology is required. Therefore, in this study, we developed a remote-controlled integrated decontamination process that simultaneously implements laser decontamination and in-situ contamination measurement that can reduce the amount of radioactive concrete waste generated during decommissioning and ensure the safety of workers.

The remote-controlled integrated decontamination process is a process in which four unit technologies such as autonomous driving, in-situ measurement, fiber laser scabbling and dust collection are integrated. Workers can remotely operate the devices to remove radioactive concrete surface contamination from outside the work space.

Autonomous driving technology measures the location coordinates of a space by applying 'Simultaneous Localization And Mapping' technology, and enables remote control to the operator as it can check the positions of landmarks and devices in the work space. In-situ measurement technology measures the concrete surface contamination degree in real time by applying the Scan survey method, not the Smear method. The Smear method has a disadvantage in that it takes a lot of time for sample collection and analysis when measuring a large-area concrete. However, in-situ measurement can quickly measure the contamination degree of the concrete surface in real time, thereby drastically reducing the working time. In addition, the contaminated concrete spot can be visualized by combining the location coordinates obtained from autonomous driving with the contamination data, ensuring operator convenience. Fiber laser scabbling decontamination technology has superior transmission characteristics, non-contact characteristics, and less energy loss compared to existing solid/gas lasers, and can significantly reduce the amount of secondary waste compared to high-pressure water or chemical decontamination. The dust collection technology collects concrete dust generated when scabbling and can prevent secondary contamination from radionuclides contained in the dust.

The integrated decontamination process developed through this study removes contaminated concrete surfaces through remote control, thereby reducing the amount of radioactive concrete waste generated and ensuring the safety of workers in the highly radioactivity work environment. However, in order to optimize the integrated decontamination process, continuous research such as technical verification and review of the applicability of the NPP decommissioning site by establishing devices to which the integrated decontamination process is applied is expected to be necessary.

Acknowledgements

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Keywords: Concrete decontamination, Remote control, Fiber laser scabbling, In-situ measurement

Development of Treatment Process for Liquid Waste of Complex-fluid Decontamination Technology Containing Nanoparticles

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In the process of nuclear power plant operation/dismantling, a large amount of various radioactive metal waste is generated depending on the purpose of use, the state of pollution, and whether to treat it. However, there is no clear treatment solution to radioactive metal waste in Korea, increasing the need to develop technologies to reduce radioactive metal waste. Wet chemical decontamination technology is used for existing decontamination system and surface decontamination among radioactive metal waste treatment technologies, which have the disadvantage of generating large amount of secondary liquid waste after decontamination. Complex-fluid decontamination technology containing nanoparticles has produced foam decontamination, reducing the amount of decontamination compared to conventional chemical decontamination technology, which compensates for the disadvantages of existing chemical decontamination technology by minimizing secondary waste. However, liquid waste of complex-fluid decontamination containing nanoparticles after decontamination was included strong acidity and surfactants, and if it enters the radioactive liquid waste disposal system in a unique state, it causes nuclide volatility and corrosion in the system by acidity. In addition, if surfactant is applied to an ion exchange resin used to remove nuclides, the fouling phenomenon reduces the exchange capacity of the ion exchange resin, resulting in an increase in the amount of ion exchange resin used. Therefore, this study developed 'Treatment Process for liquid waste of complex-fluid decontamination technology containerizing nanoparticles' to reduce the amount of radioactive metal waste generated during operation/dismantling of nuclear power plants and minimize secondary waste.

Treatment Process for liquid waste of complex-fluid decontamination technology containing nanoparticles consists of an inorganic removal process and a surfactant adsorption process in a decontamination waste solution. The inorganic removal process in the decontamination liquid waste neutralizes the decontamination liquid waste indicating strong acidity by using $Ba(OH)_2$, an alkaline chemical, and precipitates and removes radioactive metal ions present in the liquid waste. The surfactant adsorption process uses Powder activated carbon (PAC), which has a large non-surface area and is highly hydrophobic, to adsorb then remove organic-based surfactant contained in the decontamination liquid waste.

In order to evaluate the developed Treatment Process for decontamination liquid waste, an evaluation of the removal rate of inorganic matter and organic matter in the decontaminated liquid waste was performed. In case of inorganic matter removal process using $Ba(OH)_2$, the removal rate was evaluated for metal ions (Co) and as a result, removal rate was confirmed 94.85%. In the case of surfactant adsorption process using PAC, removal rate was confirmed 98.29% by deriving optimal conditions.

The optimal Treatment Process for decontaminated liquid waste was established that reduce the amount of secondary waste generation. If additional technical verification and on-site applicability review are conducted, it is deemed that it can be used sufficiently in the waste treatment process for nuclear power plants, not only in the decontamination liquid waste treatment process.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT: Ministry of Science and ICT) (No. 2020M2A8A507995221).

Keywords: Decommissioning, Radioactive metal waste, Surface contamination, Decontamination, Radioactive liquid waste

Development of Remote-Control Robot Technology for High Level Radioactive Spent Filters

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The spent filters generated by nuclear power plants are difficult to handle due to the high-level radioactivity, so they cannot be packaged in a 200 L steel drum and are stored in an enclosed space (filter room) for a long time. As the decommissioning of Kori unit 1 will be started in a few years, Radioactive wastes stored in a nuclear power plant shall be disposed of in advance so that the decommissioning project can proceed in timely. Therefore, it is necessary to develop the technology to transfer and dispose of high level radioactive spent filters stored in nuclear power plants.

In this study, we will develop a remote-control robot that handles spent filters in a high-level radioactive environment, pack spent filters into 200 L steel drum, and establish a nuclide inventory evaluation plan to ensure suitability for disposal. Through this study, we will be able to obtain techniques of radiation dose measurement and automatic classification in high-level radioactive environment using a remote-control robot. It is also expected to reduce the radiation exposure of workers.

Keywords: Spent filters, High activity radioactive waste, Remote-control robot

Evaporation and Thermal Decomposition of Paraffin From Paraffin Wasteform Including Radioactive Boric Acid Waste

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In the past, high-fidelity materials were manufactured by using paraffin, a type of soft high-fidelity medium, as a measure to treat concentrated boric acid waste. In the case of paraffin solids, about 75 wt% concentrated boric acid waste powder and about 25 wt% paraffin were mixed and are currently stored in a 200 L steel drum. However, in the case of paraffin solids, the physical chemical durability is low, so it is difficult to dispose of them because they do not meet the acceptance criteria for permanent disposal site, and it is necessary to develop treatment methods to solve them. In this study, we intend to evaluate the feasibility of separating paraffin and boric acid waste powder from paraffin solids by using the pyrolysis distillation method. Paraffin is oxidized according to temperature and decomposes into fatty acid, alcohols, esters, and ketrone. Volatile in the form of reactants and paraffin in the high temperature zone can be separated from the boric acid waste powder, and volatile components can be collected and recovered in the low temperature zone. It is expected that less than 1% of paraffin will remain in the boric acid waste powder, which eliminates paraffin, so it is believed that it can be disposed of using the existing method of treating boric acid concentrated waste.

Keywords: Boric acid waste, Paraffin, Evaporation, Pyrolysis distillation method

Development of the Glass Wool Radioactive Waste Treatment Technology

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The glass wool (insulation) generated during the replacement of steam generators at nuclear power plants is contaminated with radioactive nuclides during long-term use and classified as very low-level radioactive waste and shall be disposed of in a permanent disposal site. However, the glass wool is hardened over a long period of use and easily changed into particles by moisture contact, which do not meet the criteria for acceptance of disposal site, and technology development is needed to solve this problem. In this paper, we intend to evaluate the feasibility of how the insulations can be handled effectively. We intend to investigate the radiochemical and physical properties of the insulations and analyze technical strengths and weaknesses by classifying appropriate treatment methods to deal with them and derive basic characteristic data on them. In particular, the compression process can be considered as the simplest and most effective method, and the compression ratio can be obtained around 30%. In addition, there is a method to strengthen the physical characteristics of the glass wool by recombining silanol on the surface of the glass wool into Si-O-SI by heat treatment and chemical treatment, but this requires additional process cost compared to the compression process.

Keywords: Glass wool, Insulation, Silanol, Moisture

Simulation for Predicting the Optimal Conditions of Radionuclide Barrier Function of High Integrity Containers in Disposal Environment

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High activity homogeneous wastes (spent resin, concentrate, etc.) generated from nuclear power plants can be disposed of in the form of solidified bodies or high integrity container packages that meet the acceptance criteria for disposal. Polymer Concrete High Integrity Container (PC-HIC) was developed through localization research, however, in the process of technical explanation for obtaining permission to use it, the need for complementary evaluation on the safety of disposal was raised in relation to the duration period of HIC's radionuclide barrier function and the possibility that the barrier function would completely collapse at one moment. It is because the disposal safety can't be guaranteed if the barrier function of the HIC packaged with high radioactive waste containing a large amount of long half-life radionuclides collapses at one moment, even though HIC is designed to prevent the leakage of radionuclides in the package to the environment by maintaining integrity in the disposal environment for more than 300 years. Currently, the evaluation of the disposal safety in Wolsong rock cavern type (the first stage) repository is carried out from a conservative point of view, assuming that all radionuclides in HICs are released at one moment after 300 years of disposal. Therefore, the nuclide-specific concentration of the disposal limit should be conservatively set to be low, which makes it difficult to operate the repository efficiently. As a countermeasure, a simulation model to predict the optimal condition of long-term radionuclide barrier function of HICs was set up using variables such as the longest period during which HICs can fully maintain the barrier function, the time from the point when the barrier function begins to deteriorate to the point when all nuclides are completely released, the half-life and the allowable concentration for self-disposal of radionuclides, and the leakage rate. We predicted the duration period of barrier function and the limit of leakage rate required for HICs with the types and behavioral characteristics of radionuclides with the simulation model. It is expected that this will be able to be used as reference data for long-term integrity evaluation study on the radionuclide barrier function of HICs.

Keywords: High integrity container, Disposal safety, Long-term radionuclide barrier, Simulation model

Development of Technology for Treatment of Non-degradable Radioactive Waste

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Unsatisfactory radioactive waste generated from operating nuclear power plants is mostly generated during facility inspection, maintenance, and facility management in the management area. The type of waste is a hazardous chemical substance, which is temporarily stored in the form of sludge or liquid, and due to insufficient storage space due to the increase in the amount of untreated waste stored. It is necessary to reduce radioactive waste and develop treatment technology through the development of treatment technology for waste. Overseas radioactive waste treatment business is led by companies that have secured high technology and reliability in connection with the dismantling of nuclear power plants.

In developed countries of nuclear power plants, treatment and disposal technologies are being developed by reflecting the disposal and regulatory environment. In Korea, the need for technology development suitable for treatment, disposal and regulatory environments is recognized, and treatment, disposal safety evaluation, and related infrastructure are established. If this technology is developed, it is possible to timely treat waste that is being stored for a long time, and it can be expected to reduce disposal costs due to reduction of use. Workers' industrial safety can be secured by stable management of radioactive waste, and social acceptance for waste treatment can be secured. It is expected that it will be possible to enter the overseas radioactive waste treatment market through technological advancement in the future.

Keywords: Non-degradable radioactive waste

Development of Small Capacity 100 kW Torch Technology

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Technology development is needed to stably treat and immobilize various forms of radioactive waste (asbestos, paint, etc.) that are difficult to dispose of and dispose of during the operation and dismantling of nuclear power plants. Plasma torch melting technology reduces volume by using high-temperature heat (about 1,600℃) generated by torch using electric arc phenomenon such as thunder to ensure proper disposal. This equipment is used to generate high-temperature heat capable of thermally decomposing and melting a small amount of waste and to stably supply a melting heat source for volume reduction treatment of waste. Torch operation can be used in non-transfer, transfer and mixed modes, and is designed to prevent damage from high temperature atmosphere (1,300℃ or more) and to prevent arcing on the surface of the torch. The torch electrode material is easily replaced by an Oxygen free copper. The inside of the torch has a water-cooled structure that can be cooled by cooling water, and it can operate normally even with a maximum output of 100 kW.

KHNP CRI's PTM System dramatically reduces the volume of radwaste and the CRI provided it for working design of a facility to treat radwaste from decommissioned Kori Unit 1 and when the equipment is built, it will minimize radwaste generated during operations as well as decommissioning.

Keywords: 100 kW Torch technology, Plasma torch melting technology

Hybrid Discharge Device for Vitrification Facility

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Since the mid-1990s, the amount of waste has been reduced through the improvement of the radioactive waste management system and the development of new treatment technologies generated during operation of nuclear power plants. There is a need to develop a technology that can stabilize radioactive waste so that it does not leak into the environment while reducing the volume of radioactive waste. The vitrification technology is an induction heating type low-temperature melting method. When a glass is placed in a melting furnace and a high-frequency current is passed through the induction coil, the glass is melted by the high-temperature heat to form a molten metal.

It is a technology that traps radioactive materials as chemical defects in a stable glass structure after putting waste and pyrolysis. In this study, a device for periodically discharging molten hot glass was developed. Various methods are applied to the glass discharge technology, but the tilting method is most commonly applied. For stable discharge of high-temperature glass containing radioactive substances. A hybrid type glass discharge device was developed in which a heating device and a sliding gate valve are combined. The titanium nozzle is heated up to 1000 using a heating device to discharge the glass. The molten glass was discharged smoothly, and the glass was discharged by opening the gate valve. When the glass discharge is completed, the water cooling valve is closed to complete the discharge operation. In addition, the entire process contributed to the stable operation of the facility through real-time monitoring through CCTV. This technology is expected to be of great help in reducing the amount of generation and stably disposing of radioactive waste by vitrifying it in the future.

Keywords: Hybrid discharge device, Vitrification technology

Evaluation of the Gas Generation From Disposed Spent Resin in PC-HIC

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Polymer concrete high integrity container (PC-HIC) is being developed for efficient disposal of spent resin. An important task of the PC-HIC development is the evaluation of gas generation from radiolysis and container corrosion as these may affect the safety by increasing atmospheric pressure and increasing concentration of flammable hydrogen gas within the disposal facility. In this paper, based on expected spent resin volume as described in Hanbit Unit 3&4 FSAR, hydrogen gas generation from dried spent resin within PC-HIC in disposal environment is evaluated.

PC-HIC consists of polyethylene inner wall, polymer concrete (PC) and carbon steel outer wall. According to the acceptance criteria of the disposal facility, free water content within the PC-HIC should be below 1%, thus it conservatively assumed that PC-HIC contains spent resin and 1% of free water. Polyethylene, PC and waste are expected to be affected by radiolysis, while carbon steel is affected by corrosion.

Gas generation by radiolysis depends on dose and G-Value. Based on NRC Waste Form Technical Position Revision 1, conservatively assumed dose over 300 years is 10^9 rad, so same dose is applied for evaluation. G-Value for dried spent resin is about 0.0261 and free water is about 0.45. Thus, total gas generated over 300 years is estimated to be about 30.14 m^3.

Polyethylene with G-Value of about 0.3 would generate 0.31 m^3 per PC-HIC. For PC, the only composition that is affected by radiolysis is polyester resin, so weight averaged G-Value of polyester resin could be used to represent G-Value of PC. As G-Value of polyester resin is about 0.75, weight averaged G-Value for PC would be 0.0343. Therefore, about 0.68 m^3 of gas would be generated per PC-HIC.

Carbon steel corrosion generates negligible amount of carbon dioxide, so only hydrogen gas generation is considered. PC-HIC would be disposed in the silo with fillers, so poor air circulation is expected, resulting anaerobic corrosion. Assuming a continuous corrosion, annual anaerobic corrosion rate is about 0.005 μm, so gas generated from corrosion over 300 years is estimated to be 0.05 m^3 per PC-HIC.

Assuming 90% of PC-HIC is filled, 124 PC-HIC would be required to dispose spent resin from Hanbit Unit 3&4. Therefore, total gas generated over 300 years would be 159.27 m^3.

In this paper, a preliminary evaluation of gas generation from disposed spent resin in PC-HIC is done considering all possible gas sources, and this evaluation is planned to be done for other NPPs in future.

Keywords: Spent Resin, High Integrity Container, Gas Generation, Radiolysis, Corrosion

Preliminary Studies on Evaluation Techniques of Neutron Activation in the PET Cyclotron for Proper Planning the Decommissioning Using Monte Carlo Code

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In worldwide tens of thousands of units of particle accelerators are used for dedicated medical of commercial applications producing positron-emitting radionuclides mainly ^{18}F which has longer half-life than other radionuclides such as ^{11}C, ^{13}N, ^{15}O, etc. On the other hand, it is of note that cyclotron facilities activated by secondary neutron generated by $^{18}O(p,n)^{18}F$ reaction, which is the main process to manufacturing ^{18}F.

Evaluation of neutron activation of cyclotron facilities using computational codes is common methodology for assessing radioactive waste inventory and radiological impacts to plan proper decommissioning. In practice, several previous studies for planning decommissioning strategies of actual facilities have suggested using Monte Carlo codes (e.g. MCNP, FLUKA, PHITS, etc.) and/or Bateman equation analysis codes (e.g. ORIGEN series, FISPACT, DCHAN-SP, etc.) to calculate neutron transport and activation for structural materials of cyclotron facilities.

Because of the objectives of prior studies which were focused on each actual cyclotron facility, however, there is not enough of consolidated information about very essential input factors such as physics model or cross-section library for neutron yield calculation from $^{18}O(p,n)^{18}F$ reaction, neutron energy spectra and cross-section libraries for neutron activation calculation, impurity content of structural materials, average beam current, and depth profiles of neutron spectra of cyclotron facilities.

In this study, the impacts of each input factor for evaluation of neutron activation on specific activity of activation products were analyzed using MCNP6 and FISPACT-II. For sensitivity analysis of each input factor, a semiempirical 13-MeV PET cyclotron facility was modeled with the cyclotron body surrounded by concrete vault wall. Independent modelling with FLUKA code was conducted for validation of code modelling and results.

Neutron yield from $^{18}O(p,n)^{18}F$ reaction has two tendencies: higher neutron flux at lower energy with Bertini, ISABEL, and INCL4 physics model, and, higher neutron flux at higher energy with CEM03.03 model and TENDL-2019 library. Also, LANL 66-group cross-section library, which has specified lower energy grouping, shows higher activation products from (n,γ) reaction while 175-group VITAMIN-J shows higher ^{54}Mn specific activity because of specified higher energy grouping. If there are no impurities in structural materials, it could be possible to manage every radioactive waste from decommissioning as clearance waste while more than 15 years are needed if there are Co, Cs or Eu as reported levels in previous studies.

The results of this study can be used for proper decommissioning plan include considerations of neutron activation evaluation and radioactive waste management of cyclotron facilities.

Keywords: Cyclotron, Neutron Activation, Monte Carlo, Decommissioning, Radioactive Waste Management

A Study on Application of High Integrity Container for Boron Concentrates

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The high integrity container (HIC) is defined as a container commonly designed to meet the structural stability requirements of the US NRC document and to meet Department of Transportation (DOT) requirements for a Type A package in the United States. In Korea, the HIC is defined as a waste package that maintains its integrity more than 300 years under Korean generic underground environment and disposal condition. The target waste of HIC are spent resin, boron concentrates, etc.

The boron concentrates are dried using concentrate waste drying system (CWDS) in Korean nuclear power plants (NPP). The dried boron concentrates are coarse and fine power form. Since they are classified as a particulate with a various diameter, they should be treated to be non-dispersible. The representative treatment methods are cementation, polymerization, and HIC packaging. Previously, solidification using cement or polymer were widely used. However, some inconvenience of the methods hesitates the application of the techniques. The HIC, which alleviate the regulation and WAC, allows the more convenient application of boron concentrates disposal, have been continuously investigated.

In this study, the safe disposal of boron concentrates using the HIC is studied. The waste acceptance criteria of KORAD, international lesson learned, and current status of the HIC will be reported.

Keywords: High Integrity Container, Boron Concentrates, Radioactive Waste, Disposal, Treatment

Establishment of Optimal Quantitative Conditions for Chelate Analysis of Radioactive DAW Samples Using HPLC

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In the conventional analysis method using UV-visible, EDTA and NTA could not be quantified separately. However, when using an HPLC instrument, their retention time in the column is different, so each of them can be quantified. In this experiment, optimal quantitative conditions were established using HPLC for analysis of chelates (EDTA, NTA, Citric Acid) in the DAW.

A reverse phase ion pair chromatography column with a mobile phase consisting of 0.6 mM HNO_3, 2.6 mM tetrabutylammonium hydroxide, 7.53 mM tetrabutylammonium hydrogen sulphate and 37 μM iron chloride Anhydrous (III) was used as stationary phase and analyzed by direct measurement of UV absorption at 260 nm. The retention times of EDTA and NTA were measured at 6 minutes and 7.3 minutes, respectively.

In the case of Citric Acid, detection can be confirmed at a UV wavelength of 210 nm. For clear detection of Organic Acids, the mobile phase is detected using 0.2% Phosphoric Acid. Since various Organic Acids were detected at the UV wavelength of 210 nm, the content of Citric Acid was analyzed using the retention time of HPLC. Citric aicd's retention time is 13.2 minutes.

The chelate content of the analyzed 9 DWA samples was less than the limit in accordance with Article 16 (Hazardous Substances) of the Nuclear Safety and Security Commission's Notice (Rules for Waste Acceptance Criteria of Low and intermediate Level Radioactive Wastes), so the regulations for delivery of disposal could be satisfied.

Keywords: Chelate Analysis, HPLC, Waste Acceptance Criteria of Low and intermediate Level

Kd Measurement for Tc(VII) in TEVA Resin and Anion Exchange Resin According to the Time of Addition of the Carriers During the Sequential Separation of ^{99}Tc-^{90}Sr-^{55}Fe-^{63}Ni in Rad-Waste

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Changes in Kd value in TEVA and Anion Exchange resin according to the time of addition of carriers (before and after the separation of Tc) when performing continuous sequential separation of ^{99}Tc-^{90}Sr-^{55}Fe-^{63}Ni was identified through the batch test of the simulated sample.

The value of Kd [Tc(VII)] obtained with TEVA resin could not be observed with specific changes depending on the time of addition of the carriers, but when using an anion exchange resin, the Kd [Tc (VII)] value was 1.5 times higher in the absence of a carrier.

From the above results, it is well understood why carriers of Sr, Fe, and Ni should be added just after Tc-99 separation step in the sequential separation process of ^{99}Tc-^{90}Sr-^{55}Fe-^{63}Ni.

Keywords: Kd measurement, Sequential Separation, ^{99}Tc, TEVA

Recovery Determination of Fe-55 Separation/analysis in Rad-waste Using Fe-59 as Tracer

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A large amount of Fe carrier added for measuring the recovery rate using the commonly used precipitation method or ICP increases the amount of solvent used for adsorption and desorption by increasing the amount of expensive resin, increasing the evaporation and drying time of the solvent, thereby increasing the separation time as well as it can be the biggest obstacle to the analysis of decommissioning waste, which requires analysis of a large amount of samples in a short period of time.

The purpose of this experiment is to shorten the analysis time of Fe-55 by determining a fast recovery rate with a gamma tracer (Fe-59, β-γ, $T_{1/2}$: 44.40 d) when separating Fe-55 (EC, $T_{1/2}$: 2.744 y) from radioactive waste.

After deriving the experimental correlation from the shape of the spectrum measured in the LSC due to the quenching effect of electrons in different decay methods, the amount of radioactivity due to electrons emitted when Fe-59 decay was removed from the value of the mixed radioactivity of Fe (55+59). Thus, radioactivity analysis of pure Fe-55 could be performed.

Keywords: Tracer, Carrier, Decommissioning Waste, Recovery Rate, Quenching Effect

Review of the Possibility of Disposing of High Activity Spent Resins Using Concentration Average Method

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KHNP plans to dispose of the waste resin by packing it in a polymer concrete high integrity container (PC-HIC). However, it is difficult to meet the acceptance criteria of 10 mSv for a surface dose rate when spent resins with high activity are packaged in PC-HIC.

To satisfy WAC, the thickness of the PC-HIC should be increased or the radioactivity of spent resins should be reduced after long-term storage in a temporary storage facility. However, these methods have a problem in that the packaging efficiency of PC-HIC is lowered and the disposal cost is increased, or there are problems of securing a site and increasing management costs during long-term storage.

In this paper, the possibility of applying Concentration Averaging (CA) proposed by NRC as a method for disposing of spent resins with high dose rate stored in the storage tank of domestic NPP was examined.

CA is a method that allows low-level wastes with different radionuclide concentrations to be mixed for disposal so that the nuclide concentrations of the mixed waste can be used to classify the waste. According to the NRC's CA BTP, CA without blending is possible for single waste streams in the same package, but for multiple waste streams, it is necessary to check whether the threshold of 2015 CA BTP Table 1 is exceeded. If the threshold is not exceeded, it can be disposed of without blending, but if it is exceeded, the waste must be blended and proven to be properly mixed.

The domestic NPPs store high and low activity spent resins separately from each other, except for Kori and Hanul Units 1 and 2 (Below Kori and Hanul). According to the CA BTP, since Kori and Hanul stored spent resins with different streams in a mixed storage tank, it is necessary to investigate whether the threshold of 2015 CA BTP Table 1 is exceeded.

In the case of Kori and Hanul, where high and low activity spent resins are already mixed, the disposable mixing ratio cannot be determined. However, since other NPPs store high activity and low activity waste resins separately, if they are to be packaged in PC-HIC, they can be mixed by determining a disposable mixing ratio.

In this paper, the possibility of disposing of high activity waste resins stored in domestic NPPs by 2015 CA BTP. In the future, it is planned to calculate the disposable mixing ratio according to the concentration of spent resins.

Keywords: Spent Resin, Concentration Averaging, PC-HIC, Blending

Evaluation of Radiation Shielding Performance for Uranium Material Storage Container Using Microshield

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The Korea Atomic Energy Research Institute is operating a nuclear material storage facility to store materials generated by experiments and managing a variety of nuclear materials. Some of these nuclear materials are being stored in large quantities and should be checked to ensure proper shielding performance for the safety of workers. The nuclear material to be evaluated is stored in containers with about 12 tons of depleted uranium with a U-235/U-238 ratio less than 0.72%. A storage facility is a place where nuclear material is only stored, so materials cannot be opened. Therefore, internal exposure was not considered, but only external exposure was considered. The assessment was made using the shield performance evaluation code microshield, and the input factor (radiation per unit weight) was calculated by referring to the health physics and radiologic health handbook. The dose assessment library utilized ICRP 107. The source is being stored in cylinder form, the shielding material is steel (ASME SA516 Grade 70), and the shielding thickness was evaluated from 0 mm to 15 mm at intervals of 5 mm. As a result of the evaluation, cylinder's maximum radiation dose rate is 2.71E-01 mSv/h at a distance of 10cm and it decreases by an average of 3.92% with an additional 1mm thickness. At a 1m distance, maximum dose rate is 9.84E-02 mSv/h and it decreases by an average of 2.94% with an additional 1mm thickness. As a result of the screening performance evaluation, it was confirmed that the thickness of the shield body currently in use is 159 mm, which is sufficiently safe compared to the worker exposure standard (20 mSv/yr). These findings will be used to assess doses to other nuclear material in storage.

Keywords: Depleted uranium container, Shielding performance, Microshield, Worker exposure rate

Determination of the Density Correction Factor for Samples Analysis of Clearance Wastes in Gamma Spectroscopy

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At nuclear power plants, gamma spectroscopy was used to mainly evaluate for concentration in a Reactor Coolant System. Therefore, conventionally, the radioactivity of a sample generated during operation has been analyzed by applying energy and efficiency calibration using a certified reference material of 1g/cc, which is the density of coolant. However, as decommissioning of nuclear power plants begins, clearance wastes having various densities are being generated. Although there is a difference in density between the calibration source and the sample of clearance waste, the radioactivity has been analyzed by applying the calibration efficiency of the existing 1g/cc source. Therefore, the objective of this study is to calculate the density correction factor and analyze the radioactivity by applying different efficiencies according to the density of the sample.

To determine the density correction factor, the simulated efficiency of the Eu-152 point source was calculated by modeling the HPGe detectors with 40% relative efficiency of *CANBERRA* and *ORTEC* using the MCNP code. To demonstrate the validity of the modeling, the measured efficiencies for the Eu-152 point source were obtained for each HPGe detectors, and then compared with the simulated efficiency. As a result of comparison, it was confirmed that the relative error was within about 5%, and the validity of the MCNP code was verified. To determine the density correction factor, six certified reference material with densities ranging from 0.02 to 5.44 g/cc were used. Each source emits multiple energy lines ranging from 88 to 1836 keV. The efficiency of each energy line for each density was measured using HPGe detector, and then compared with the MCNP simulated efficiency of the same geometry, respectively.

As a result of comparing simulated efficiencies and measurement efficiencies, it was confirmed that the relative error was within 10% in all energy lines, and all density correction factor were determined using the results of MCNP code. The density correction factor was calculated as the ratio of the efficiency of each density to the efficiency of the 1g/cc source at the same energy. In addition, to expand the density range of measurable samples, each efficiency curve according to the density from 0.00 to 5.44 g/cc was obtained using the log fitting function. The results of this study may contribute to improving the reliability of evaluation of radionuclide inventory for clearance wastes in the future. Further studies will also be carried out to develop a program to automatically apply the corrected density-specific efficiency.

Acknowledgements

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Keywords: Density Correction Factor, Clearance Waste, Radionuclide Inventory, MCNP code

Application of RESRAD-OFFSITE to IAEA Vault Test Case for Radwaste Disposal Safety Assessment

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RESRAD-OFFSITE code, one of the RESRAD (RESidual RADioactivity) family code, has been developed by ANL for evaluating the radiological doses and cancer risks to a human receptor who located inside (onsite) or outside(offsite) of the primary contamination area. Recently, RESRAD-OFFSITE is being implemented to the safety assessment of radioactive waste disposal facilities such as in IAEA Biospheric Model Validation Study (BIOMOVS) II and Environmental Modeling for Radiation Safety (EMRAS) Naturally Occurring Radioactive Material (NORM) working group study, etc.

In order to investigate the applicability of RESRAD-OFFSITE to the post-closure safety assessment of engineered vault type LILW (Low- and Intermediate- Level Waste) disposal facilities, benchmarking exercises were, in this paper, conducted to IAEA ISAM Vault Test Case and the assessment results were compared with those of the previous study. In the previous study, AMBER compartment modeling approach was implemented to assessment of the liquid release pathway well scenario. Among the new source term models in RESRAD-OFFSITE version 4 which was released in 2020, two types of the source release models, exponential release model and equilibrium desorption release model, were used to calculate the radionuclide concentrations of well water. The concentrations of well water which was 300 meters away from outside the area of disposal vaults were compared with those from AMBER model. From the comparison of peak concentrations and their timings for key radionuclides (C-14, Tc-99, I-129, U-234, U-238) in the well water, it can be seen that there is a good agreement (always within an order of magnitude) between RESRAD-OFFSITE with equilibrium desorption release model and AMBER calculations, although different approaches in source term release and groundwater transport modelings were implemented in two codes. Comparing both of the RESRAD-OFFSITE source term release models, exponential release model gives lower concentrations than equilibrium desorption release model.

The comparison of calculation results showed that RESRAD-OFFSITE could appropriately simulate the radionuclide release from engineered vault type disposal facilities. Furthermore, considering the releases of radionuclides from the primary contamination to the atmosphere, surface runoff and groundwater can be simulated by the code, it is expected that RESRAD-OFFSITE could also be implemented to the post-closure safety assessment of VLLW (Very Low Level Waste) disposal facilities.

Keywords: RESRAD-OFFSITE, Safety Assessment, IAEA Vault Test Case, Disposal

Evaluation of Radioactive Waste Characteristics in PC-HIC

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Concentrate waste and spent resin from nuclear power plants will be planned to be loaded in PC-HIC (Polymer Concrete-High Integrity Container). PC-HIC review report from KORAD states that waste with a maximum external surface dose rate of 10 mSv/hr of PC-HIC can be disposed and that it can be transported to the Gyeongju disposal facility if the external surface dose rate of PC-HIC is less than 2 mSv/hr. Considering 2.6 times of the shielding effect of PC-HIC, radioactive waste up to 26 mSv/hr can be disposed. With these criteria, the evaluation of waste characteristics was carried out by ERIR (Evaluation of Radioisotope Inventory of Radioactive waste) data base of KHNP (as of December 2019).

Before operating SRDS (Spent Resin Drying System), spent resin was stored after cement solidification. After operating SRDS, it has been dried and stored in the HIC. Considering the surface dose rate of spent resin, storage drums were classified as 38.37% for those below 2 mSv/hr, 8.31% for 2~10 mSv/hr, 9.51% for 10~26 mSv/hr, 27.51% for 26~150 mSv/hr and 16.31% for those above 150 mSv/hr.

Before operating CWDS (Concentrate Waste Drying System), concentrate waste was stored after cement solidification. After operating CWDS, it was processed in paraffin solidification and stored. Recently, concentrate waste has been dried and stored in powder form. Considering the surface dose rate of concentrate waste, storage drums were classified as 84.19% for those below 2 mSv/hr, 13.86% for 2~10 mSv/hr, 1.73% for 10~26 mSv/hr, 0.22% for 26~150 mSv/hr.

Radioactive waste with surface dose rate of below 26 mSv/hr(spent resin: about 56%, concentrate waste: about 99.8%) is not required to be shielded, but the radioactive waste over 26 mSv/hr can be disposed to disposal facility when acceptance requirements are satisfied, considering additional shielding strategy.

Keywords: PC-HIC, Spent Resin, Concentrate Waste, Surface Dose Rate

Zeolite-Kapok Membrane for Cesium Adsorption and High Volume-Reduction of Radioactive Liquid Wastes

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Zeolite-containing kapok fiber membranes were developed to remove radionuclides from radioactive liquid wastes and reduce the total waste volume. Kapok fiber has high hollow ratio and combustion characteristics burning out easily at low temperatures without generating ashes. In this research, various kapok-zeolite composite membranes were manufactured and compared by utilizing several types of zeolite and fabrication treatments. Based on the basic comparison tests, a wet-laid nonwoven manufacturing process was used for the large-area continuous membrane fabrication, and the manufacturing conditions were optimized to increase the adsorption by zeolite and the volume reduction of kapok fibers during combustion. The hydrophilic treatment process was established in which the lumen of kapok fibers were maintained and lignin was adequately left for high burning rate. The combining ratio of materials, the roller speed, and the compressed force affect the adsorption and burning rates of the composite membrane. When measured by TGA in a nitrogen environment, it showed a combustion rate of over 95% for 2 hours at 8℃/min heating rate. The adsorption experiments of radioactive cesium (Cs-137) were conducted with the fabricated membrane samples of 150 cm^2 in the very low-level radioactive solution. The final adsorption rate becomes over 90% within 15 minutes in the experiments. The biosorption technology without the secondary waste volume could be applied for large-capacity, low-level liquid wastes such as radioactive water tanks in Fukushima, Japan.

Keywords: Radioactive waste, Volume reduction, Zeolite, Kapok, Cesium, Adsorption

Reinforce on Leaching Resistance of Solidified Cement by Hot Isostatic Press

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The disposal issue about radioactive waste is important in terms of safety ensure safety public. Most of solid type radioactive waste can be compacted, melted, or solidified entirely. However, because radioactive sludge generated from contaminated tank in nuclear power plant is not solid type and consists of water and powder including various oxide material, it should be solidified for disposal.

Solidification of radioactive sludge is efficient disposal method, but radioactive waste volume is increased by solidifying additive material. And then, when self-solidification is conducted using only radioactive sludge with solidifying additive materials such as metal, cement, ceramic, compressive strength and leaching resistance of those solidified radioactive sludge can be lower than solidified radioactive sludge using solidifying additive materials. Therefore, reducing the quantity of solidifying additive materials and satisfying leachability index and compressive strength for the standard of solidified radioactive waste disposal at the same time. Leachability index can be analyzed by inductively coupled plasma mass spectroscopy (ICP-MS) device.

In this paper, several sludges were encapsulated by solidified cement. The volume of solidified cement capsule is same, but the wall thickness of solidified cement capsule is different. It means that when the volume of solidified cement capsule is same, the thicker wall thickness of solidified cement capsule, the smaller quantity of sludge can be disposed in solidified cement capsule. When the wall thickness of solidified cement capsule is thinner, although the quantity of sludge can be disposed in solidified cement capsule can be increased, leachability index and compressive strength of capsule can be degraded.

In this process, it is expected that reprocessing of solidified cement capsule in high temperature and high pressure by hot isostatic press (HIP) will enhance the leachability index and compressive strength. Considering the generation of crack on cement surface in the temperature higher than 400℃, temperature in HIP was lower than 400℃. Pressure in HIP was controlled from 3000 PSI to 5000 PSI. By physical and chemical enhancement of solidified cement capsule in HIP process, it is a research key to find an optimal wall thickness that satisfies the disposal criteria for leaching resistance and compressive strength while disposing as much of the sludge as possible in the same volume.

Keywords: Solidified Cement, Radioactive Sludge, Cobalt, Leachability Index

5분과

제염해체 (Oral)

In-Situ Sr-90 Analysis in Soil From Selective Gating of Energy Channels on PVT Probe

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Traditionally, quantitative analysis technique for Sr-90 radioactivity concentration in environmental soil requires sampling and radiochemical separation of Sr-89/Sr-90 and Sr-90/Y-90 nuclides in laboratory, which is time-consuming work. As an alternative way, in this study, we suggested and tested a rapid analysis method of Sr-90 radioactivity concentration in environmental surface soil by using a 12 mm (Φ) × 20 mm (H) size polyvinyltoluene (PVT) probe coupled with a 12 mm (Φ) × 100 mm (H) poly (methyl methacrylate) (PMMA) light guide. Here, the counts from the probe are gained by a photomultiplier tube and discriminated in 1024 energy channels.

Because Sr-90/Y-90 decay energy shows 2,279 keV at maximum, which is higher energy than those of other major radionuclides generated from nuclear facilities or naturally, counts in proper energy channels (corresponded to energy region from 1,470 keV to 2,279 keV) were selectively chosen to characterize Sr-90 on site rather than usage of counts in full energy channels for gross beta analysis. Specifically, detection efficiency was compared and calibrated for each beta and gamma radionuclide. After the calibration, this method was verified by measuring fabricated soil samples, hardened by flocculation and mixed with liquid Sr-90 sources to have 1 Bq/g, 3 Bq/g, and 10 Bq/g concentrations. As a result, in case of gated energy channels, accuracy for Sr-90 concentration estimation was even higher than that of gross channels at 1500 s; the relative standard deviations of gross and gated mass efficiency values for three concentrations were 14.5% and 3.8%, respectively. Measurement time for less than 900 s was enough to satisfy the minimum detectable concentration of 1 Bq/g, the lowest range value for clearance level in solid materials. If this technique is used in combination with the other scanning devices such as gamma spectrometer, effective sites survey would be possible in actual decommissioning cases.

Keywords: In-situ analysis, Sr-90, Gross beta, Energy channel, PVT probe

Chemical Stability Study for SAP(SiO_2-Al_2O_3-P_2O_5) Wasteform in Domestic Underground Waster

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Metal chloride waste generated from the pyroprocessing is one of the most problematic wastes when solidifying radioactive waste to reprocess used nuclear fuel. Since Chloride waste has a chemical stability issue, we chose vitrification after dechlorination process rather than direct vitrification. Upgraded SAP (SiO_2-Al_2O_3-P_2O_5) was developed by adding solidification promoters such as Boron and Aluminum to SAP which was developed as a composite for indirect solidification after dechlorination. Previous studies have identified the physical properties, structure and chemical stability of U-SAP. The Product Consistenct test (PCT) is currently used to test the form of glass and glass ceramic waste in hazardous waste. It is mainly used to compare the chemical stability of glass substances with another glasses. However, there are limitations to investigating the long-term chemical stability of a substance. To investigate the leaching behavior of U-SAP with monolithic morphology, durability test C1308 was conducted with solidified specimens. The ASTM C1308 test method provides a procedure to measure the leaching rate of an element in its solidified form. In this article, a leaching test was conducted for a total of 56 days using a specimen of U-SAP material obtained by dechlorination and solidification with surrogate salt. Deionized water and KURT ground water (reducing condition, 500 m deep underground) was used as leaching solution, and the test was conducted at 20, 40, 70℃. The leaching data resulting from ion migration on the surface was obtained through the ICP-MS and ICP-OES and the normalized leaching rate was calculated. The alteration layer was observed through SEM and SEM-EDAX, and the composition change on the alteration layer was also observed. This result will be useful in evaluating the long-term stability and chemical durability of U-SAP waste solidified material.

Keywords: Alteration layer, Upgraded SAP(SiO_2-Al_2O_3-P_2O_5), ASTM C1308, Chemical stability

Choline Chloride-Based Deep Eutectic Solvents as an Alternative Decontamination Agent for Radioactive Metal Waste

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Deep eutectic solvents (DES) are a mixture of a hydrogen bond acceptor (HBA), commonly quaternary ammonium salt, and a hydrogen bond donor (HBD), such as a carboxylic acid or alcohol. These solvents are readily synthesized from easily accessible materials. DES are characterized by their low cost, melting point, flammability, toxicity and wide electrochemical windows. Chemical and physical properties of DES can be tuned through judicious choice of HBA and HBD such as acidic HBD's which readily dissolves metal oxides. This illustrates novel advantages over currently used or proposed decontamination agents such as ionic liquid, molten salts or strong acids.

In this research, we tested the feasibility of DES synthesized from choline chloride (ChCl) and p-toluenesulfonic acid (PtsA) as a decontamination agent. ChCl:PtsA was selected owing to its high metal oxide solubilizing power. Metal oxides such as Fe_3O_4, CoO, Cr_2O_3, and NiO present in the contaminated layer of stainless steel 304 were studied and were shown to have good solubilities in ChCl:PtsA.

To simulate decontamination process, simulant contaminated stainless specimens were produced. Stainless steel 304 samples were oxidized at 800°C for 30 minutes with constant steam supply and were cooled gradually in the furnace. The formation of layer was evident from SEM images, and the layer composition was studied by SEM-EDS. Laser-induced breakdown spectroscopy (LIBS) data was collected for oxide layer, which showed peaks that correspond to metals aforementioned.

The simulant contaminated stainless specimens were leached, and their oxide layers were successfully removed in ChCl:PtsA. ChCl:PtsA solvent leaching kinetics and final concentration of metals were investigated using ICP-OES. SEM-EDS and LIBS results of specimen surfaces pre-leaching and post-leaching were compared to assess decontamination ability. SEM images showed that the oxide layer was efficiently removed after the leaching. LIBS data of specimen pre-oxidation and post-leaching were almost identical, which shows that the metal oxide layer was removed and the base metal was exposed. These results corroborate that the eutectic mixture formed from two biocompatible components has an excellent solubilizing power which efficiently dissolves and removes oxidation layer from the contaminated stainless steel.

This work proposes and proves the possibility of using ChCl:PtsA as a decontamination agent. It was shown that while ChCl:PtsA with good metal oxide solubilizing power has numerous advantages over conventional leaching agents, it was also able to decontaminate simulant contaminated steels. The efficiency of the process is expected to be further improved when electrochemical process is combined.

Keywords: Deep eutectic solvents (DES), Decontamination, Solvometa llurgy, Metal recycling, Green solvents

Assessment of Impact on the Statistical Test of MARSSIM due to the Residual Contamination Distributions of Survey Units Using Random Sampling Method

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After permanent shutdown of nuclear power plants, the sites have been remained and their residual radioactivity affected surrounding environment. Currently, Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) is widely used for site release. In MARSSIM, nonparametric statistical methods have been used to determine whether the average radioactivity concentration of survey units exceeds Derived concentration guideline level (DCGL) in the final status survey (FSS). However, it has been reported that the actual environmental samples and the radionuclides concentration in the soil have specific distributions such as lognormal distribution. In addition, studies performing nonparametric tests of MARSSIM assuming survey units with specific distributions of radioactivity have been rarely reported. It is hard to determine whether to release survey units before performing FSS. If the possibility of site release before FSS can be predictable, there are some advantages such as performing preemptive decontamination if necessary. Therefore, this study has generated populations with certain distributions, established a random sampling model and performed the statistical test used in MARSSIM to compare statistical power for the purpose of predicting the release probability of survey units with specific distributions.

Using Oracle Crystal Ball tool for Monte Carlo simulation, lognormal distribution, normal distribution, max extreme distribution, min extreme distribution and uniform distribution have been generated. The location parameters such as mean of assumed distributions were set to 50 respectively and the contamination range from 0 to 100. Total 81 cases have been set by varying the scale parameter, relative shift, type Ⅰ decision error (α) and type Ⅱ decision error (β) to independent variables and 10,000 times of random sampling have been performed for each simulation case.

The DCGLs of each simulation case satisfying the designed decision errors by counting the number of samplings which reject the null hypothesis by performing Sign test have been evaluated. have been derived and statistical characteristics of each distribution have been analyzed. Moreover, some issues for each distribution have been found which were important in terms of decision errors. This study can be used to support preliminary decision whether to release the site with a specific distribution.

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Keywords: MARSSIM, Nonparametric statistical test, Lognormal distribution, Normal distribution, Random sampling, Oracle Crystal Ball

Decontamination Performance Test by Using a Laser for Radioactive Contaminants of Various Materials and Shapes

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Various materials and shapes of radioactive waste are generated in domestic nuclear energy utilization facilities. Radioactive waste is classified into surface contamination and activation according to the location and distribution of radioactive contamination. Surface contamination is classified into partial surface contamination and total surface contamination depending on the contaminated area.

In this study, a decontamination performance test for surface contamination by material and shape of decontamination targets was conducted using a laser. Epoxy coating, stainless steel and carbon steel were tested as a material-specific tests, and flat surface, curved surface and bolt were tested as a shape-specific tests. Performance tests were conducted with laser powers of 100W and 500W, respectively.

If decontamination can be achieved by overcoming the material and shape constraints of radioactive waste generated from nuclear energy utilization facilities, the amount of self-disposal can be increased. As a result, the two benefits will be obtained: cost reduction for permanent disposal and resource recycling.

Compared to conventional decontamination methods such as abrasive, chemical or ultrasonic decontamination, the decontamination performance test using a laser by material and shape has the following advantages in terms of decontamination effect and time. (1) As it is easy to decontaminate only contaminated areas, it is possible to minimize the generation of secondary waste, (2) decontamination is very easy even for complex shapes, and (3) most materials generated in nuclear energy utilization facilities can be decontaminated. Therefore, it is expected to be useful in nuclear power plants, medical institutions, research institutes, general industries, etc.

Keywords: Nuclear energy utilization facility, Radioactive waste, Surface contamination, Laser, Decontamination

Behaviors of Metals in a Chelate-free Inorganic Chemical Decontamination Process

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Structure materials undergo corrosion in a high temperature and high pressure condition during the operation of nuclear power plants, and the corrosion products are transported to the core where they are activated by neutron. The activated corrosion products deposited on the structure materials, and they cause several problems such as decrease of the heat transfer efficiency and build-up of the radiation fields in the reactor coolant system. To resolve these problems, various chemical decontamination technologies using organic acids have been applied to remove the activated corrosion products from the reactor coolant system. KAERI has been developing a chemical decontamination process using the chelate free inorganic acid, SP-HyBRID process. It was confirmed that the SP-HyBRID process has similar decontamination performance to that of the commercial HP/CORD UV process. A Secondary waste generated from the SP-HyBRID process can be significantly reduced when compared with the HP/CORD UV process. In this study, the behaviors of metals in the SP-HyBRID process was investigated. The concentrations of the metals were variously changed in the decontamination process, and the changes accorded with the objective of each process in the decontamination process. The metals were removed effectively from the decontamination wastewater in the wastewater treatment process.

Keywords: HyBRID, Decontamination, Metal ion, Dissolution

Investigation of Radioactive Waste Level of Channel Structures in Decommissioning Wolseong Unit 1

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Wolseong Unit 1, a pressurized heavy water reactor (PHWR), is being prepared for the world's first PHWR decommissioning. PHWR fuel assemblies are loaded into pressure tubes (PT) which penetrate the length of the reactor vessel (called the calandria). The PT and calandria tubes (CT) should be operated with low power, as the cooling water flow is restricted with sagging due to irradiation creep and growth during plant operation. To overcome this degradation, PT and CT of Wolseong Unit 1 were replaced which had been in operation for 25 EFPYs, but the plant was permanently shut down 2 EFPY after the replacement. In this study, in order to establish the decommissioning plan of Wolseong Unit 1, radioactive waste level for CT/PT before and after replacement was evaluated. The level is determined by activities per unit mass of each nuclide, which was obtained by performing MCNP and ORIGEN computational analysis. 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 a result, PT was evaluated as ILW in both cases, whereas CT as ILW when operated for 25 EFPYs, and LLW 2 EFPYs. The neutron flux and energy spectrum in PT and CT are almost similar due to their adjacency, and chemical compositions of the PT and CT are composed of more than 97% zirconium. Since the PT material contains 2.5% Nb, it is classified as ILW even after 2 EFPY operation by the reaction of Nb-93(n,gamma)Nb-94. On the other hand, the reaction of Ni-62(n, gamma) Ni-63 in CT containing less than 0.1% Ni is the main reaction, so it shoud be operated for more than 10 years to be classified as ILW. This study shows how important the effect of small amounts of elements in materials is in determining the radioactive waste level.

Keywords: Wolseong Unit 1, Channel Structures, Neutron induced Activation, Radioactive Waste Level

Verification Test of the Kori Unit 1 Dismantling Technologies Using the Mockup

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The Reactor Pressure Vessel (RPV), one of the relatively highly activated structures in the nuclear power plant of Kori unit 1, should be taken care of when it is dismantled from within containment building. The related technologies are needed for RPV dismantling and developed technologies need to be verified. In the case of reactor pressure vessel of Kori unit 1 with relatively low activated by neutron irradiation than internal structure, we suggest a method of in-situ remote dismantling by combining thermal cutting while lifting by strand jack systems from the current location. The reactor inlet/outlet nozzles and the insulation module structures are separated by a mechanical cutting method during the preparatory works, and the reactor shell is cut by an oxygen-propane method while lifting, and the lower head is placed on a basket-shaped fixture pedestal and cut out using a plasma cutting method. When the nuclear reactor is dismantled, a ventilation equipment is operated to collect radioactive pollutants in the workplace. The cut-off pieces are cut by the cutting plan, and the cut-off pieces are moved and placed in the packaging containers by the optimal loading plans. Finally, after the remaining insulation module structures are removed and all residual radioactive waste cleaning is done, and then the entire reactor pressure vessel dismantling process is finished. We made RPV cutting or dismantling devices to implement the developed process and performed each verification test. Furthermore, in order to optimize the process of in-situ dismantling the RPV in Kori unit 1 containment building, we simulated and performed verification test using 100%-sized demonstration mockup. In addition, exposure was evaluated using MCNP and radiation protection plan was established based on the dismantling process of the Kori unit 1 nuclear power plant. We calculated the working hours and estimated total amount of radioactive waste based on the results of empirical tests for Kori unit 1 RPV dismantling process. In this paper, we introduce the overall process of in-situ dismantling the reactor pressure vessel and the results of verification tests for them.

Acknowledgements

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Keywords: Reactor Pressure Vessel, In-situ, Dismantling, Pilot test, Radioactive Waste, Mockup

Development of Tritium (^{3}H, T) Removal Technology From Liquid Radioactive Waste Containing Tritiated Water (HTO)

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The radionuclides that are released into the environment with the operation of a nuclear power plant and have the greatest impact on the environment are tritium (^{3}H, T) and ^{14}C. As ^{3}H was detected in the groundwater around the Wolsong nuclear power plant, where three heavy water reactors are currently in operation, concerns and anxiety about the internal exposure of residents around the power plant amplified. As a result, social interest in ^{3}H removal technology is increasing rapidly, and great interest is also being gathered in the emission management of these nuclides. Commercially available technologies for ^{3}H removal so far include LPCE (Liquid Phase Catalytic Exchange) technology used in domestic heavy water reactor called TRF (Tritium Removal Facility) and CECE (Combined Electrolysis Catalytic Exchange) technology used at the Fukushima in Japan. Because these technologies use cryogenic catalytic exchange and/or electrolysis, their processing capacity is limited, and their high operating costs are accompanied by economic problems. Therefore, the development of large-capacity ^{3}H removal technology is urgent.

To solve the above problems, this research focuses on the development of new high-capacity/high-efficiency technologies for ^{3}H removal by improving and convergence of previously published 4 different type of ^{3}H removal technologies. Currently, after analyzing and grasping the characteristics of existing technologies, research and improvement on these technologies are being conducted with four participating universities. Among the four HTO separation/removal technologies, the first technology is a vacuum-pressure process using a surface-modified porous inorganic adsorbent, which is a vapor phase HTO removal technology. This technology is currently in the process of developing the adsorbent and has an HTO removal efficiency of about 20%. The second technology is a liquid phase HTO removal technology using an ion exchange membrane and lithium manganese oxide (LMO), which is known as a ^{3}H adsorbent. From the results of the experiments so far, it is judged that the possibility of commercialization of this technology is slim, so we are focusing on developing HTO separation technology through the convergence of electrochemical modules and LMOs. This technology is still being researched, and now it has achieved a ^{3}H removal efficiency of about 15%. The third is a liquid HTO removal technology through an ion exchange resin column. Scale-up research is in progress along with research on reproducibility of the existing technology for this technology. Zeolite membrane technology is also considered as an alternative technology for ^{3}H removal from contaminated water. With this technology, about 20% of ^{3}H removal efficiency was achieved. Currently, this research is being conducted to increase the ^{3}H removal efficiency of zeolite membrane and to increase capacity by synthesizing the membrane on a support with a wide permeable area. In the future, we plan to construct a semi-pilot scale ^{3}H removal system to clearly identify the characteristics of each improved technology and develop a hybrid type ^{3}H removal system capable of large-capacity processing. In this paper, the research results of each technologies being developed and the strategy to develop a hybrid type ^{3}H removal system by combining these technologies were briefly introduced.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 20191510301170).

Keywords: Tritium, Tritiated water, Liquid radioactive waste, Separation technology, Isotope, Nuclear Power plant

Design of the Chamber for Mechanical Decontamination of the Waste Electric Lines by Process of Nuclear Power Plant Decommissioning

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There are various types of electric lines used for purposes such as power, lighting, control, and instrumentation for the operation of nuclear power plants. However, when decommissioning a nuclear power plant, it is difficult to find a way to separate, treat and recycle these electric wires. In particular, the recycling of wires is the most desirable direction, but it requires careful consideration of the degree of radioactive contamination and appearance. As a condition for recycling of waste wires, first of all, the wires must have a good surface condition for visual identification. And then, radioactive particles must be completely removed from the wire surface by dry, wet and mechanical decontamination methods. Whereas the dry or wet method is to wash contaminant particles directly through the fluid, mechanical decontamination is to use friction to remove particles that have stuck to the surface of the wire that cannot be resolved with the previous method. So, mechanical decontamination can be seen as one of the last measures to determine the recycling of waste wires. In this study, we designed the shape of the chamber for mechanical decontamination, and focused on selecting the location of the gas inlet and outlet so that decontaminated particles in the chamber can be discharged smoothly without stacking. As a result of analysis using a commercial CFD tool, it has been discovered that the chamber size and gas pressure drop between the entrance and exit of the chamber are highly dependent on the initial conditions of the particles to be mechanically decontaminated. And if the particle initial velocity is very slow and the particle size is as small as micrometers, we confirmed that decontaminated particles in the chamber can be effectively discharged by selecting an appropriate gas inlet and outlet location even with a small gas pressure drop of 1kpa or less.

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Keywords: Decommissioning, Electric Lines, Mechanical Decontamination, NPP

Sequential Treatment Process for the Volume Reduction of Radioactive Concrete Waste From Decommissioning Projects

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Concrete used as the main material for buildings and structures is a mixture of aggregate and hydrated cement. Large volumes of radioactive concrete wastes are generated during the decommissioning of nuclear facilities. This concrete waste has a huge impact on the decommissioning cost, because the cost of disposal of decommissioning wastes is dominant portion for the total budget. Radionuclides in the contaminated concrete waste were mainly existed in the cement components, due to the high adsorption properties of porous cement. Therefore, the volume of radioactive concrete waste could be effectively reduced by separating non-contaminated aggregate fraction from the cement component. In this study, thermomechanical and chemical treatment conditions were optimized for the aggregate to meet the clearance criteria by using simulated samples contaminated with non-radioactive and radioactive Cs and Co. As a result, more than 70% volume of the waste could be separated for clearance by the sequential treatment process. Therefore, it is expected that the disposal cost could be dramatically decreased, and also enhanced utilization of the disposal space could be realized.

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Keywords: Decommissioning, Radioactive concrete waste, Volume reduction, Thermomechanical treatment, Chemical treatment

Development of Heterostructure P-doped Co_9S_8@MoS_2 for Efficient Tri-butyl Phosphate Degradation

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Tri-butyl phosphate (TBP) is one of the organophosphorus compound with high production quantities of 3000-5000 tons/yr and wide industrial applications for flame retardant, plasticizers, herbicides, and anti-foaming agent. Specifically, TBPs have commonly employed as organic solvent for separating uranium in PUREX process in the nuclear industry. However, TBP itself is also a persistent contaminant for a long period after disposal, causing environmental problems. For example, in groundwater near legacy radioactive site, the presence of organic chelate like TBP has potential to facilitate the mobility of waste component. Considerable levels of TBPs can exist for several decades without natural degradation, enhancing mobility of the radionuclides and pollutant into the surrounding environments. To reduce its potential hazard and toxicity for human beings, the researchers have applied various approaches (i.e. incineration, biodegradation, and Fenton reaction) for TBP degradation. Among these techniques, photo-oxidation method seems to be one of the effective ways because of some advantages including simple reaction route, cost effectiveness, and safety remediation to the environment. Several promising studies shows possibilities of TBP degradation using UV light. Nevertheless, their photocatalytic application is very restricted by the UV zone, which accounts for only 5% of the solar spectrum.

Molybdenum disulfide (MoS_2), a subclass of transition-metal dichalcogenides (TMDs), is emerging 2-dimensional semiconductor material. In particular, many researchers have focused on its unique optical property to harvest visible-light. However, still MoS_2 has limitations like low conductivity, fast recombination and dominant basal planes, which is relatively inert than active sites. Among the various modification methods recently have been adapted, construction of heterojunction has emerged as new strategy, attributed to the synergistic effect of combining MoS_2 with other conductive material. A series of visible-light catalyst (i.e., C_3N_4, Ni_3S_2, CoS_2, Co_9S_8)-MoS_2 hetero-composite has attracted much attention with great chemical properties.

By inspired by previous studies, we synthesized P-doped Co_9S_8@MoS_2 hetero-composite by a facile two-step process, a hydrothermal synthesis followed by phosphor-sulfidation process. Synthesized composites exhibited 95.82% of TBP degradation efficiency within 3hr under visible light irradiation. Compared with pristine MoS_2 and Co_9S_8, Co_9S_8@MoS_2 showed improved photocatalytic activity due to the larger surface area, and construction of heterostructure interface between Co_9S_8 and MoS_2. In addition, about 3 wt% of phosphorus doping into the Co_9S_8@MoS_2 showed enhanced catalytic performance. This work demonstrated a new strategy to degrade TBP under visible light irradiation with great efficiency.

Keywords: TBP degradation, photocatalysis, Visible light irradiation, P doping, Heterocatalyst, Metal sulfide

Indirect Radionuclide Inventory Assessment for Decommissioning Radioactive Wastes: Current Status and Future Prospect of Scaling Factor Method

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Radioactivity of radionuclides in radioactive wastes must be identified to ensure the safety of the repository in accordance with the Nuclear Safety and Security Commission's notice. The radioacitvity of gamma-emitting radionuclides can be easily detected by means of non-destructive method from outside of the waste packages. However, in order to meausre that of beta and/or alpha-emitting radionuclides, which is called difficult-to-measured (DTM) nuclides, it requires time-consuming and labor intensive destructive radiochemicaly analysis. Performing radiochemical analysis on all radioactive wastes is not pratical, indriect radionculdie assessment methods has been adopted. Of the methods, scaling factor (SF) method, which predicts the radioactivity of DTM nuclides based on the correlation between DTM nuclides and key nuclides, has been used as the principal method in many countries. Approximately 6,200 tons of decommissioning radioactive wastes expected to be generated for 900-1,300 MWe pressurized water reactor. In order to dispose of such large amounts of decommissioning radioactive wastes, indirect methods such as SF method are essential. Although SF method has been implemented in many countries, based on our current knowledge, it is applied only for nuclear power plant’s operational wastes. For decommissioning waste, some problems are expected to arise that make it difficult to apply existing SF method. There is one international standard of SF method proposed by International Organization for Standardization (ISO) but it is not sufficient to solve the problems due to the lack of details. In the present work, we reviewed international experiences of SF implementation and proposed some potential issues to develop reasonable and advanced statistical decision-making. It is expected to be helpful for the development of advanced indirect radionuclide inventory assessment method for nuclear decommissioning wastes as the new era of nuclear decommissioning begins.

Keywords: Scaling factor, Decommissioning radioactive waste, Radionuclide inventory, Decision criteria, Statistics

Calcium-based Solution as Washing Agent for the Removal of Strontium in a Soil Near Nuclear Power Plant

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Soil in the vicinity of a nuclear power plant is likely to be contaminated with various heavy metals including nuclide such as strontium (Sr). Even for non-radioactive Sr, it can be dangerous to human body because of its physicochemical similarity to calcium (Ca). In this study, taking advantage of this similarity, calcium was introduced as solute for novel washing agents for the removal of strontium in field soil in the vicinity of a nuclear power plants. For comparison, magnesium ion was also tested for the removal of strontium. Washing condition was determined from our previous study, determined by Box-Behnken design, consisting of concentration of 1 M, L/S ratio of 20, washing time of 1 h, and pH of 2. As a result, calcium and magnesium-based solutions showed relatively low removal efficiency (lower than 20%) for anionic heavy metals (Cr and As), and higher removal efficiency (mostly about 30 to 50%) for cationic heavy metals (Sr, Cs, Co, Cu, Pb, and Zn). It was noticeable that the removal efficiency of Sr by calcium-based solution showed outstanding value than other results, showing 68.2%. Calcium-based washing agent tends to exhibit higher removal efficiency depending on similarity between ion and target heavy metal. Moreover, a five-step sequential extraction method was employed to before and after washing with solution, in order to figure out which chemical form of Sr was mainly liberated from soil in the course of Ca-based solution washing. It was found that regardless of bound forms, either weak or strong, Sr was extensively released. Furthermore, commonly used washing agents with their typical concentration (i.e., 0.075 M EDTA, 0.01 M citric acid, 0.01 M oxalic acid, and 0.05 M phosphoric acid) were selected and tested for comparison with ion-based washing agents. EDTA, citric acid, oxalic acid, and phosphoric acid exhibited Sr removal efficiency between 25 to 30%. Other than Sr, other heavy metal removal efficiencies were quite similar. Additionally, unlike the other strong acid washing agents, Ca-based solution seemed to liberate Fe/Mn oxides bound Sr through an ion exchange mechanism, not by destroying the oxides themselves.

Keywords: Soil washing, Physicochemical similarity, Strontium, Calcium

Decontamination Process Optimization for CANDU Zirconium Alloy Waste

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Zirconium based alloy has been widely used in nuclear industry due to its low thermal neutron absorption, especially for nuclear fuel cladding, pressure tube, and calandria tube. They are expected to be disposed as intermediate level radioactive waste which cannot be accepted in Gyeongju LILW disposal facility. Only surface decontamination method is not enough to achieve required decontamination factor because high neutron exposure condition in reactor operation period raises amount of activation product volumetrically. In this study, electrorefining process for decontamination of Zr-based alloy adopting ZrCl recovery method and rotating cylinder electrode in 10 kg/y scale electrorefiner is investigated.

Nuclear-grade Zr-2.5Nb alloy was used as anode material, and rotating tungsten cylinder electrode was utilized as cathode in all experiments. A cathode potential of -1.1 V (vs. 1 wt% Ag/AgCl) was applied in all experiments for zirconium metal ion recovery as ZrCl. First, potentials of anode and cathode and current of cell were monitored by rotating speed to specify the electrorefining conditions, such as rotating speed and anode/cathode space area ratio. Secondly, 24h experiment of electrorefiner with specified condition was operated.

For optimized condition specification experiments, 1.61 and 5.25 of anode/cathode area ratio were compared by anode potential monitoring and rotating speeds of cathode were chosen as 0, 10, 30, 100, 300 rpm. First, in an area ratio of 1.61 case, anode potential kept rising up to -0.2 ~ -0.1 V which can cause oxidation of Nb and Ni. In an area ratio of 5.25 case, anode potential kept in range of -0.8 ~ -0.7 V which can oxidize only zirconium from Zr-2.5Nb. In optimization experiments of rotating speed, 10 rpm case have shown the maximum averaged current in among the cases kept anode potential not to exceed -0.7 V. Optimized conditions (anode/cathode surface area ratio > 5.25, 10 rpm) were applied into 10 kg/y scale electrorefiner system. Deposits were analyzed by XRD and ICP-OES. A ZrCl with adhered electrolytes were identified by XRD. ICP-OES revealed that the weight percent of Zr, Nb, and Ni have changed 99.9839%, 0.0159%, and 0.0001998%, respectively.

Keywords: Volumetric decontamination, Irradiated Zr alloy, Intermediate level radioactive waste, Electrorefining, Rotating cylinder electrode

Derived Improvements From the Experience in Development of NPP Decommissioning Waste Packaging and Transportation Containers

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The purpose of this paper is to share the gained experiences and derived improvements in the design of packaging and transport containers developed in the KETEP project, “Development of NPP Decommissioning Radioactive Waste Packaging, Transportation, and Disposal Containers”. It is summarized into four items as follows. The characteristics and specifications of each container are described in detail in the final report of Phase 1 of this project.

First, we proposed the lifting requirements, stacking requirements, and surface dose requirements as the design requirements of the packaging container in consideration of the suitability for disposal and handling. Although the drop requirement was excluded, in the performance test, the drop test was performed by deriving the height considering the collision energy that may be generated during handling. As the weight of the packaging container for decommissioning waste is expected to increase, it is necessary to establish a design standard for the packaging container considering safety and economical efficiency.

Second, Korea demands a longitudinal acceleration factor of 10 g for all transport container retention systems, while this is required by IAEA and NRC only for type B or higher transport containers. Compared to smaller design weight of the transport container, the increased the design weight to 35 tons would cause issue of a design complexity of the container and an increase of manufacturing cost. Therefore, it is recommended to amend the related notification so that the relaxed retention requirements can be applied to transport containers of type A or lower, such as IAEA and NRC.

Third, the packaging container developed in this project is made of carbon steel having a thickness of about 6 mm. However, since the width and length of the corner casting, which plays a key role in maintaining the structural integrity of the container, are each 162 mm, the entrance area is about 70% smaller than the inner area of the container. This causes the problem of not being able to utilize 100% of the internal space and consequently increases the disposal cost, so improvement is needed.

Finally, the packaging container for surface disposal facility should be optimized and designed in consideration of the size of the disposal facility and the interval between containers.

If packaging and transport containers are designed and manufactured in consideration of the experiences and improvements presented in this paper, it is expected that they will contribute to the efficient utilization of domestic disposal facilities.

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Keywords: Packaging, Transportation, Container, Retention, Acceleration factor

5분과

제염해체 (Poster)

Dead Leg Investigation in Full System Decontamination

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The full system decontamination prior to decommissioning and dismantling is generally carried out for primary systems that directly or indirectly contact a coolant containing radioactive materials.

Usually, the optimal technology and process for the decontamination target is applied, but, there are some places called 'dead legs' that do not decontaminate well. A dead leg is an area in a piping or tubing system where the decontamination agent can become stagnant and not exchanged during flushing. Dead legs include a multitude of piping configuration such as blanked branches, lines with normally closed block valves, lines with one end blanked, drains, relief valve inlet and outlet header piping, vents and instrument connections.

The important reason for investigating the dead leg is that a local hot spot exists in the areas associated with dead legs after system decontamination and release the radioactive materials into the system. KHNP is considering RCS, CVCS and RHRS as the system decontamination range of Kori Unit 1. The dead leg investigation of Kori Unit 1 was performed with reference to the dead leg criteria of Indian Point 2. The criteria of dead leg in Indian Point 2 were (1) whether the main process piping was horizontal, (2) whether the branch line tap at the bottom of the main process piping is, and (3) whether the ratio of the distance from the branch pipe tap to the primary isolation point and the diameter of the branch piping exceeds 10 times.

Based on the dead leg criteria of Indian Point 2, through P&ID, PLAN drawing and ISO drawing of Kori Unit 1, it was confirmed that there were 30 branch pipes in the RCS and 27 of them were dead leg. Currently, the dead legs of CVCS and RHRS in Kori Unit 1 are being investigated, and cleaning method and procedures to remove the radioactive materials in the dead leg area are being developed.

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Keywords: System decontamination, Dead leg, Dead leg criteria, Kori Unit 1

Decontamination of Sr Contaminated Concrete Waste via Chlorination Technique

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Upon decommissioning of nuclear power plants (NPPs), concrete waste is one of the largest volume among various radioactive wastes. Although almost of the concrete waste might be classified into low- or very-low-level radioactive waste, it is known that there also exists concrete waste contaminated by Sr-90. Due to high mobility and health issues against human body, Sr-90 is of serious concern. Therefore, additional treatment for secure disposal of Sr-90 without bulky concrete waste might be beneficial from waste management point of view.

Previous approaches normally employ acidic solutions to dissolve radioactive nuclides along with non-radioactive concrete constituents. This solution goes through step-by-step precipitation for separation. In this work, chlorination technique was introduced as a new means of decontamination of Sr contaminated concrete waste. The key idea of this technique is to utilize chlorine gas for selective conversion of SrO into $SrCl_2$ which can be easily removed by water washing. Thermodynamic calculations were conducted to predict chlorination reaction behavior of Sr compounds as well as major constituents of concrete waste. The calculation results suggested that SrO might react with chlorine gas prior to CaO, $CaCO_3$, and MgO, while SiO_2, Al_2O_3, and Fe_2O_3 remain as their oxide forms. However, experimental results showed that the chlorination reaction should be conducted at least 700℃ to achieve more than 40% of SrO conversion into $SrCl_2$. Experimental results revealed that chlorination reaction of Ca compounds was unavoidable during the chlorination reaction of SrO.

Keywords: Concrete waste, strontium-90, Chlorination reaction, Decontamination, Decommissioning

Assessment and Analysis on Radiation Exposure Dose in Operators During a Decontamination Process of Pressurizer of Pressurized Heavy Water Reactor

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Dismantling reactors, steam generators, and pressurizers at a nuclear power plant requires detailed strategies as they are massive structures in size. Especially, a pressurizer is directly connected to the primary side (water chamber, tube) of the steam generator. Therefore, a high concentration of radiation is expected to be activated, but detailed plans for its dismantlement are insufficient compared to other structures. This study estimated a radiation exposure dose among workers working on the dismantlement operation of a large structure of pressurizer. As an evaluation tool, VISIPLAN Software was employed, which was actually applied to the dismantlement of BR-3. Oxyacetylene Torch (8 Segments) was adopted as the pressurizer dismantlement method, and the operation duration was set at 8.625 hr/man. This information was derived from NUREG/CR-3587 for reference for this study. Furthermore, due the difficulties in securing the radiological data of pressurizers, the radionuclide inventory data (600MW(e) CANDU - Immediately after reactor shutdown) were adopted for this study. This reflects the pressurizer's characteristic of being directly linked to the primary side, where most of the contamination is incurred in the steam generator. It was assumed that the dismantlement operation took place one meter away from the pertinent structure, and it was cut at the several different sections: inlet, bottom, five sections on the body, and top. The evaluation of the mentioned parts produced the following values respectively: 3.6E+02, 8.7E+01, 2.3E+01, 2.1E+01, 2.1E+01, and 2.2E+01, 5.6E+01 mSv. However, these values were generated without the general decontamination process. Therefore, they were re-evaluated based on the decontamination factor (DF) of 10, which was achievable in case the chemical decontamination process was carried out in the water chamber of the steam generator. As a result, they were estimated at 3.6E+01, 8.7E+00, 2.3E+00, 2.2E+00, 2.2E+00, 2.3E+00, and 5.7E+00 mSv respectively. This is averaged at 9.5% when compared to 100 mSv as the maximum limit of the annual effective dose for workers.

The results of this study did not reflect a contamination distribution per normal water level of the pressurizer but it was assumed to be entirely contaminated, making them difficult to be applied to an actual on-site case. Therefore, future studies are expected to reconstruct a contamination distribution according to normal water levels of pressurizer and conduct a dosimetry based on detailed and various operation scenarios.

Acknowledgements

This work was supported by a Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korean government (MOTIE) (20191510301120, Development of Industrial Technology & Standards for Radiological Characteristics Phase of Permanent Shutdown NPPs).

Keywords: Pressurizer, Dismantling, Radiation exposure, VISIPLAN, Effective dose

Improved Underwater Laser Cutting for Dismantling Nuclear Facilities

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The cutting technology for dismantling nuclear facilities requires cutting in thick metal as well as underwater cutting by remote control. Laser cutting is the best tool to meet these requirements. Therefore, technology development for applying laser cutting to the dismantling of nuclear facilities has been actively carried out. And successful underwater laser cutting results have been reported.

However, when cutting thick steel plates of 50 mm or more, the cutting speed had to be limited to a low speed of ~5 mm/min for the initial section of 10-15 mm. If the cutting at a higher speed, the initial cutting failed. The reason is that it takes a lot of time for the target material to heat up to the melting point as the laser beam meets the thick part immediately after the start of the cutting process. It was expected that oblique cutting could improve the initial cutting performance by giving the effect of gradually increasing the cutting thickness.

In this work, a study was conducted on the improvement of the initial underwater laser cutting performance for thick steel plates though oblique cutting. As a result of the cutting tests, the initial cutting speed for stainless steel and carbon steel plates increased significantly at the oblique angle of 15°. For stainless steel plates, the maximum initial cutting speeds were 100, 50, and 15 mm/min for thicknesses of 48, 57, and 68 mm, respectively. For carbon steel plates, the maximum initial cutting speeds were 40, 15, and 7 mm/min for thicknesses of 49, 59, and 69 mm, respectively.

Acknowledgements

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Keywords: Laser cutting, Laser processing, Underwater cutting, Nuclear decommissioning, Nuclear dismantling

Considerations in View of Safety Regulation for Melting Processing Technology of Radioactive Metal Waste During Decommissioning of a Nuclear Facility

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The waste compositions have a great effect on the choice of thermal technologies, as well as the nominal capacity of the machine, the construction material, and the design of the off-gas cleaning system. Each waste to be treated has a certain specific activity for beta, alpha and gamma emitters. Any thermal treatment system will require licensing or permitting from a regulatory authority. When selecting a thermal technology, one should first invest an appropriate amount of time investigating the existing operational experience within the nuclear industry and other industries.

The melting process consists of preparation, melting, sampling and analysis, conditioning of secondary waste, and recycling, decaying or final storage of metal product. In metal melting, the waste feed material, usually scrap metal, is fed into a furnace. Additives may be used to improve slag separation and trapping of certain radionuclides. After melting, the slag is removed, samples for determination of the remaining content of radioactivity are taken, and the molten metal is poured out into a solid form. During the melting process different elements and their radioactive isotopes are redistributed between the slag, the metal, and the off-gases depending on elemental properties. The more volatile elements such as cesium, iodine or hydrogen leave the melt and are essentially transferred to the off-gases or, in some cases, to the slag. Some metallic elements such as cobalt, nickel, chromium, iron, zinc and manganese mainly remain within the melt. Transuranic elements can be readily oxidized and will transfer to the slag.

A melting facility for radioactive scrap metal is normally to the scale of a small foundry. It is important to optimize the size of the furnaces. A small furnace requires more preparation work. A larger furnace requires a higher investment and larger melt batches. Based upon the experience, the furnace should have a typical capacity of steel. Metal melting a high cost technology from life cycle perspective. mobile melting facility is typically not economical. However, the pre-treatment steps such as segmentation and decontamination can be mobile. Large commercial melting facilities are available internationally and transport of metal to these facilities for processing is an option. Licensing should not be a major issue, since all major risks for a properly designed facility are the same as for conventional foundries. The environmental impact from a melting facility with a state of the art off-gas filtration system is very limited. The positive environmental impact is that metals can be recycled.

Keywords: Decommissioning, Melting technology, Metal waste, Radioactive waste

A Study on Estimation of Radiation Exposure Dose During Dismantling of RCS Piping in Decommissioning Nuclear Power Plant

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In the dismantling process of RCS piping, a radiation protection plan should be in place to minimize the radiation exposure dose of dismantling workers. Hence, is necessary to estimate the individual effective dose in the RCS piping dismantling process when decommissioning a nuclear power plant. In this study, the radiation exposure dose of dismantling workers at different positions was estimated using the MicroShield dose assessment program. The individual effective dose, i.e., the sum of the effective dose for each tissue considering the working time, was used to estimate the radiation exposure dose. The detailed dismantling process of RCS piping was analyzed based on the NUREG-1595 report. The radioactivity of the surface contamination of RCS piping was calculated by considering the decontamination factor (DF) and half-life of the ^{60}Co radionuclide. Finally, the individual effective dose in the dismantling process was estimated by considering the radiation exposure reduction methods. The estimations of the simulation results for all RCS piping dismantling tasks satisfied the dose limits prescribed by the ICRP60 report. In dismantling the RCS piping of the Kori-1 or Wolsong-1 units in South Korea, the estimation and reduction method for the radiation exposure dose, and the simulated results of this study can be used to implement radiation safety for optimal dismantling by providing the radiation exposure dose of dismantling workers.

Keywords: Individual effective dose, Decommissioning, RCS piping, MicroShield program, NUREG-1595, ICRP60

A Study on Performance Comparison of In-Situ γ-ray Detectors for Discrimination of γ-ray Peaks Induced by Neutron Activation in Decommissioning of NPP: Monte Carlo Simulation Study

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A large amount of the metal and the concrete radioactive waste are generated during decommissioning of the nuclear power plant (NPP). Especially, the neutron activation cause high levels of activity in the concrete, and the large components that are part of the primary circuit of the NPP. Based on the concentration of each radionuclide, the levels of the radioactive waste are determined to select the disposal method. Hence to reduce the cost of radioactive wastes, it is important to discriminate and analyze the peaks of the γ-ray induced by neutron activation. The purpose of this study is the performance comparison of in-situ γ-ray detectors for the discrimination of the γ-ray peaks induced by neutron activation using MCNP6. The cadmium zinc telluride (CZT), high purity germanium (HPGe), and lanthanum bromide ($LaBr_3$:Ce) are currently available as the γ-ray detector in decommissioning of NPP. Because the discrimination of the γ-ray peaks from the energy spectrum of the metal and the concrete depends on the energy resolution of the radiation detectors, a Gaussian energy broadening (GEB) for each detector was applied for realistic simulation. Moreover, to rapidly and easily identify the radionuclide of the neutron activation, the sensitive nonlinear iterative peak clipping (SNIP) algorithm was applied to obtain energy spectrum. The results of simulation provide that in-situ γ-ray detectors are suitable for the analysis for the radionuclide of the neutron activation in decommissioning of NPP.

Keywords: γ-nuclides of neutron activation, Cadmium zinc telluride (CZT), High purity germanium (HPGe), Lanthanum bromide ($LaBr_3$:Ce), MCNP6, Sensitive nonlinear iterative peak clipping (SNIP)

Monitoring the Distribution of γ-rays for Region of Hot-spot in Decommissioning of NPP Using Cubic Compton Camera: Monte Carlo Simulation Study

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To protect dismantling workers from the radiation exposure dose during decommissioning, it is necessary to monitor the distribution of the γ-rays for the hot spots. In conventional γ-cameras, the distribution of the γ-rays cannot be directly reconstructed for other directions because their imaging coverage is limited to the direction of incident radiation. However, since a cubic Compton camera (CCC) with 4π field-of-view can detect radiation from all direction, it can replace multiple conventional γ-cameras resulting high detection efficiency and low cost. In this study, the γ-ray imaging system using CCC was proposed, and its performance was evaluated for the main γ-nuclides (^{137}Cs and ^{60}Co). The entire imaging system forms a cubic structure that generates images on the basis of radiation interactions from every direction. Hence the radiation incident at any direction was clearly reconstructed by CCC. Lanthanum bromide ($LaBr_3$:Ce) scintillators were chosen as the detector material due to their high energy and timing resolution, availability in large volume, and usability without cooling equipment. The list mode maximum likelihood expectation maximization (L-MLEM) method was used for the reconstructed Compton images. The simulation code was MCNP6 and every photon was tracked using a PTRAC card which is a built-in function in the simulation code. The results of this study demonstrate the availability of using CCC as a means of monitoring the position and distribution of main γ-nuclides in decommissioning of NPP.

Keywords: Cubic Compton camera (CCC), γ-nuclides (^{137}Cs and ^{60}Co), Lanthanum bromide ($LaBr_3$:Ce), MCNP6, Decommissioning

Study on Solidification of a CO_2 Mineralization Product Containing C-14 by Low Temperature Sintering

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Activated carbon wastes from the air cleaning system in a nuclear plant contain a long half-life value of C-14. The concentration of C-14 in the activated carbon wastes exceeds the clearance level criteria (1 Bq/g). Studies on the removal of C-14 from the activated carbon wastes have been actively conducted to meet the criteria for the self-disposal. The removal of C-14 can be effectively conducted in a thermochemical process, where a CO_2 mineralization product containing C-14 is generated. This mineralization product must be fabricated into a stable form for the final disposal.

The CO_2 mineralization product is in a form of alkaline earth carbonate, and it can be decomposed into an alkaline earth oxide and CO_2 gas at a temperature higher than 600℃. This means that the mineralization product must be fabricated into a waste form at a temperature below 600℃.

In this study, fabrication tests of $CaCO_3$ into a waste form were conducted using a bismuth oxide glass with a low melting point. The $CaCO_3$ was fabricated into a homogeneous glass-ceramic waste form with a high density (> 3.0 g/cm^3). It was confirmed that the waste form had a high chemical-durability through leaching tests (ANS 16.1, PCT).

Keywords: Activated carbon waste, CO_2 mineralization product, C-14, Waste form

A Basic Study on Immobilization of a Secondary Waste From a Chemical Decontamination (SP-HyBRID)

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SP-HyBRID (Sulfuric acid Permanganate Hydrazine Based Reductive metal Ion Decontamination) is a chemical decontamination method for the reactor coolant system in a nuclear power plant, and it has similar performance to that the commercial process (HP/CORD UV). Secondary waste generation from the SP-HyBRID process can be significantly reduced when compared the commercial process. The waste from the SP-HyBRID process is $BaSO_4$ waste powder containing metal hydroxides. These hydroxides contain radioactive metals. These $BaSO_4$ waste thus must be immobilized into a stable waste form for final disposal.

In this study, the immobilization of a simulated SP-HyBRID waste containing simulant radioactive cobalt into a waste form was conducted using a sintering method. The simulated waste was immobilized into a monolith waste form without the SO_2 gas generation. This waste form has a high density and a good compressive strength. It was confirmed the waste form had a high chemical-durability through a semi-dynamic leaching tests.

Keywords: Immobilization, $BaSO_4$ waste powder, SP-HyBRID, Waste form, Final disposal

The Surface Modification and Characterization of SiO_2 Nanoparticles for Higher Foam Stability

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The surfactant and colloidal nanoparticles has been considered for various applications because of interaction of both complex mixtures. The hydrophilic SiO_2 nanoparticle could not be surface active behavior at the liquid/air interface. In this study, the SiO_2 nanoparticles have been modified with 3-isocyanatopropyltriethoxy-silane (ICP), and the effect of foam stability has been investigated. The physical properties of surface modified SiO_2 nanoparticle were analyzed by XRD, TGA, FT-IR, and SEM. After surface modification of SiO_2 nanoparticles, the contact angle of SiO_2 nanoparticle was also increased from 62 to 82° with increased ICP concentration. The experimental result has shown that SiO_2 nanoparticle with ICP was positive effect and improved foam stability could be obtained at proper ICP concentration compared with un-modified SiO_2 nanoparticle.

Keywords: SiO_2 nanoparticle, Foam stability, Silica functionalization

A Study on the Control of Boric Acid (Boron) in Discharge Water of Nuclear Power Plants

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In a Nuclear Power Plant (NPPs), Boric Acid is used in the primary system to control the Reactor reactivity, and the reactivity is controlled by changing the Boron Concentration. Boric Acid exists in the form of Ortho Boric Acid[$B(OH)_3$] and ionized Mono-borate Ion [$B(OH)_4^-$] in water, and the presence ration of the two chemical species depends mainly on pH. Most of the waste Boric Acid-containing water generated during the operation of the Power Plant is recycled through a recovery process, but some Boric Acid water is discharged through the ion exchange resin of the Liquid Radioactive waste treatment System (LRS). In order to increase the efficiency of removing boric acid discharged, the pH in the discharged water must be maintained at 10 or more, but the pH limit value of the discharged water of Nuclear Power Plants is between 6 and 9, so it is difficult to maintain the pH more than 10 in reality. Currently, most of the Boric Acid in the discharged water from Nuclear Power Plants is being discharged without being removed. Of course, there is currently no regulation on Boric Acid in the discharged water from Nuclear Power Plants, but the National Institute of Environmental Sciences recently designated Boric Acid (containing more than 0.3%) as a toxic substance. This means that boric acid is highly harmful to humans and the environment, and regulations on handling and use are expected to be strengthened in the future. Therefore, in this paper, various methods to minimize the influence of pH and ionize and remove Boric Acid were studied, and a Pilot-Scale Boric Acid Control Device was developed through the final verification test. The main core of this device is to ionize Boric Acid with an Oxidizing Agent (H_2O_2) and then inject Sodium Hydroxide (NaOH) to adjust the pH to increase the ionization efficiency of Boric Acid. Next, after Coagulation and Precipitation of Boron ions using a Coagulant ($CaCl_2$), the precipitation is recovered and the residual Boron ions in the supernatant is removed by using a filtration membrane (RO). As a result of the experiment, the flow rate of the final treated water was maintained above 2>Lpm, and the Boron concentration was measured to be less than about 5 ppm, satisfying the target value (100 L/hr, less than 5 ppm).

Keywords: Precipitation, Coagulation, Boron, pH, Oxidizing Agent, Coagulant

Characteristics of Waste Generated From Decontamination of Small and Medium-sized Metal Parts in NPP Decommissioning

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To provide basis for developing a decomposition process of organic waste from small and medium-sized (S&M-sized) metal decontamination in NPP decommissioning, its characteristics are set as follows by the review of published literatures.

For S&M-sized metal waste generation, an amount of 30,820 kg is estimated from Kori 1 (587 MWe) decommissioning. With this, an extrapolation, referencing to the estimates in the US and Sweden based on 1000 MWe, gives the generation as 52,000 kg/1000 MWe. Assuming that the metal waste is composed of 2 inch valves which undergo decontamination by liquid with an organic agent, it is estimated possible to treat 3,290 kg of metal waste with 1 m^3 of decontamination liquid, and wash 1,500 kg of metal waste with 0.5 m^3 of water. Applying these estimations to a 1,000 MWe NPP decommissioning, the generation of organic decontamination liquid and wash water would be respectively 15.8 m^2 and 17.3 m^2, leading to 33 m^2 in total.

For the decontamination, organic acids such as oxalic acid, citric acid, EDTA, NTA, and picolinic acid are used, and their concentration in the organic decontamination waste is found widely ranged from 10 to 1000 ppm, from which this study has set to the maximum value of 1,000 ppm. The composition of metal ions in organic decontamination waste is calculated by averaging actual decontamination data, resulting in 52.7 wt% Fe, 16.3 wt% Ni, 15.1 wt% Cr, and 15.9 wt% Mn. The average concentration of metal ions is confirmed to be 950 ppm, however this study has set a slightly higher value i.e., 1,000 ppm. Now, the major radionuclides deposited on the crud of reactor primary system are ^{60}Co, ^{58}Co, ^{54}Mn, ^{59}Fe, ^{51}Cr, and ^{137}Cs, among which ^{60}Co and ^{58}Co become main radioactivity contributors as the cooling time after NPP shutdown becomes longer. Since ^{58}Co has the same chemical properties as ^{60}Co, it is replaced with ^{60}Co. The specific activity of all activated nuclides remaining in the decontamination waste is very widely ranged. Unlike the high level contamination case such as primary steam generator tubes, the contamination level of S&M-sized parts will be quite low and, therefore, it is set to be 2,000 Bq/g. As the contribution of fission products to radiation exposure is known about 5%, and the ^{60}Co Gamma Ray Dose Constant (GRDC) is about 3.6 times higher than the ^{137}Cs GRDC, the specific radioactivity of ^{137}Cs estimated by applying these rules is 360 Bq/g.

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Keywords: Organic, Metal, Radionuclides, Waste, Review

A Case Study of Plant Zoning Experience for the Decommissioning of Nuclear Power Plants

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This study covers the initial radiological characterization, development of schedule for the activities associated with the decommissioning of the Zorita Nuclear Power Plant (NPP) in Spain. In this study, we have drawn heavily on BNFL Group experience in the planning and execution of Pressurized Water Reactor (PWR) decommissioning projects achieved in the U.S. and Europe.

The aim of this study is to present available information and lessons learned about plant zoning experience from similar decommissioning projects that may be applied to Kori unit 1 decommissioning because of the close similarity between Zorita and Kori unit 1.

The selection of the various decontamination and dismantling alternatives for Nuclear Power Plants decommissioning requires a reliable characterization data and extent of contamination at the different areas of the facility.

This characterization is based on the development of a zoning or classification scheme, requires criteria such as the expected extent of residual external (floor, walls, ceiling, external piping and equipment surfaces) and internal (internal surfaces of equipment, piping, valves and etc.) contamination level, as well as, in selected plant areas near the reactor core, the expected bulk material activation levels for the pieces of equipment located inside the room.

The rationale behind this approach is as follows. Bulk material activation of equipment parts and internal equipment contamination, mostly due to activated corrosion products, is a controlling factor for both the radiological risk expected during equipment removal operations and for the expected final radiological inventory of dismantling products. External deposited contamination on room walls and ceiling and on equipment and auxiliary structures inside them, are determinant when establishing plant decontamination strategies, aimed to reduce the expected radiological risk of dismantling operations and to minimize the amount of dismantling wastes. Therefore, this room zoning approach presents definite advantages from the classification point of view, since a rooms, normally, equipment belonging to the same system with similar contamination potential. The zoning effort at room level is performed only for those buildings with a significant radiological potential. The rest of buildings and areas are classified as non-radioactive areas.

In first place, the zoning process requires the classification of NPP different systems according to their radiological implications, both during normal operation and during permanent plant shutdown conditions.

In second place, It is necessary the establishment of room classification criteria. The proposed methodology is based on the following criteria. Presence and intensity of activated material (Metal or Concrete) inside or at room walls, presence and intensity of internally contaminated fluid systems equipment inside the room, and presence and intensity of external deposited contamination at room walls and contents.

The process delineated above has been applied to classify the room. In the case of rooms belonging to Containment, Auxiliary and Radwaste Buildings, the classification make by straightforward application of the above methodology. In the case of rooms belonging to other potentially radioactive buildings, the classification will rely on conventional considerations.

Thus, the radiological surveys require the analysis of the NPP zonification or classification scheme. This methodology can be applied to prepare for the decommissioning of Kori unit 1.

Keywords: Decommissioning, Zorita NPP, Zoning

Decontamination Method of Synthetic Tritium Solution Using Microalgae

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This study was conducted to check the tritium decontamination effect by microalgae in a monolithic synthetic radioactive solution containing tritium (H-3). Two species of microalgae (*Spirulina sp., Chlorella vulgaris*) were used in this study. We have shown that *spirulina sp., Chlorella vulgaris* have a potential to decontamination of tritium. The experiments were conducted at room temperature and adjusting pH of the solution to 5.5, taking samples at a specific time. And the tritium concentration of the solution was 2.0 Bq/mL. The microalgae need to be injected with oxygen to increase the life time, and the physical stirring method can destroy the microalgae's cells, so an aeration stirring method was used. The analysis was made using LSC (Liquid Scintillation Counting), which is effective for β-ray nuclide analysis. As a result of the experiment, the most effective reaction time was 6 hours, and at this time, Maximum tritium decontamination rate of the chlorella vulgaris, *spirulina sp.* were 16%, 28.6%, respectively.

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Keywords: Decontamination, H-3, Tritium, Microalgae, Spirulina sp., Chlorella vulgaris

Purification Test of Sr Nuclide From Salt Waste Generated From Pyrochemical Process

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The Strontium-90 nuclide is beta emitting radionuclide. It decays sequentially into Yttrium-90 and Zirconium-90 through beta decay and the heat released during beta decay is used as a source of radioisotope thermoelectric generator (RTG). Also, Sr-90 and Y-90 are used as a source of beta particles and extensively utilized in medical and industry fields. In particular, Y-90 has high beta radiation energy and short half-life, so it is used for radiation therapy. Therefore, high-purity Sr nuclides are required for the application of Sr-90 and Y-90. Sr-90 is present in a spent nuclear fuel and can be separated in a reactive-distillation process during pyroprocessing, a spent nuclear fuel recycling process developed by the Korea Atomic Energy Research Institute (KAERI), where barium is obtained together by co-precipitation due to chemical similarity. In addition, Sr-90 finally decays into Zr-90 during the storage period after separation. Therefore, Ba and Zr should be separated to obtain high purity Sr nuclides. In this work, we demonstrated that high-purity Sr can be obtained through chromate precipitation reaction, in which the pH is controlled by ammonium hydroxide, and carbonate precipitation reaction. Through the optimization study on precipitation conditions, Sr purification process was established with a high yield of over 90% and purify of over 99.9%.

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

B_4C Synthesis for Upcycling and Disposal of Radioactive Boric Acid Waste

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In NPP (nuclear power plant), boric acid is used as a neutron absorbent. So radioactive boric acid waste are generated from various waste streams such as discharge or leakage of reactor coolant water, floor drains, drainage of equipment for operation or maintenance, reactor letdown flows and etc. Boric acid liquid waste in KOREA NPP were concentrated and dried with NaOH to control the pH. Concentrated boric acid waste are stored as powder form (Borax, B_2O_3). Depending on KHNP, 20,015 drum (200 L drum) of concentrated boric acid waste were stored in KOREA NPP until 2019. For the treatment of boric acid waste, there are three major options. 1. Boric acid recovery, 2. Discharge, and 3. Immobilization. However, discharge has political and environmental problems. Recovery and immobilization options also have economical and stability problems.

In this study, our group suggests B_4C component as new options for boric acid treatment. B_4C were successfully synthesized from radioactive boric acid waste and active carbon waste. Synthesized B_4C were characterized by XRD, SEM, EDS, EA, and ICP. Physical and chemical stability of B_4C were analyzed by using TGA and PCT-A leaching test. Recycled B_4C has superior property as neuron absorbent and shows enough stability as disposal form with proper immobilization options.

Keywords: Boric acid waste, Active carbon waste, B_4C, Neutron absorber, Waste disposal

Gas Hydrate-based Decontamination of Radioactive Liquid Waste

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When the nuclear power plant is dismantled, the normal radioactive liquid waste treatment facility is not available. Therefore, in this study, a mobile radioactive liquid waste treatment platform consisting of gas hydrate, micro filtration (MF) bag filter, and fiber filter module for ion exchange is being developed for decontamination. In this paper, a method for purifying liquid waste by the gas hydrate crystallization method, which is the core process of the radioactive liquid waste treatment step, is described.

The gas hydrate crystallization is a method of separating and extracting pure water from a contaminated radioactive waste liquid by a compressed gas hydrate pelletizing reactor, and removing the remaining concentrated radionuclides including non-radioactive compounds.

As a result of the experiment, in contaminated water containing salts and radionuclides (TDS 35,000 mg/L), it was confirmed that the desalination rate was maintained over 85% based on ICP-OES. It showed also constant decontamination performance regardless of the types of elements such as Na^+, Ca^{2+}, Cl^-, SO_4^{2-}, strontium (Sr^{2+}), and cobalt (Co^{2+}). In particular, it was confirmed that this method can remove radionuclides such as cobalt and strontium, etc., of about 85-90% or more without a pretreatment process.

Acknowledgements

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Keywords: Dismantling, Radioactive liquid waste, Gas hydrate crystallization, Micro filtration, Fiber filter

Tritiated Waste Treatment Technology Development

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Tritium is a weak beta-emitting radioactive isotope of hydrogen. Tritium is produced by neutron absorption of deuterium. And tritium- deuterium can be used as a nuclear fusion fuel. Tritium has a half-life of 12.3 years, changing into helium-3. Tritium can be a weak radiation hazard when inhaled, ingested via food or water, or absorbed through the skin. The limit of the World Health Organization for tritium in drinking water is 10,000 Bq/L. Tritiated water of very low concentration is of little health hazard. The elimination of tritium from water is not easy, because the separation factor of tritiated water from light water is very low.

Research has begun to take seriously tritiated waste treatment in 1968, when the first CANDU (Canada Deuterium Uranium) nuclear power plant started the commercial operation. CANDU reactors generate tritium in their coolant and moderator, due to neutron capture in deuterium of heavy water. The presence of tritium contributes to the radiation dose of the plant personnel and radioactive emission from the reactor system. Thus Chalk River Laboratories (CRL) has reduced the operational tritium concentration through detritiation of the heavy water. CRL has developed the Combined Electrolysis and Catalytic Exchange (CECE) and liquid-phase catalytic exchange (LPCE) bed technologies.

Korea has twenty four nuclear power plant units in operation. Among the twenty four units, Wolsong units 2, 3, and 4 are of CANDU reactors. Korea Atomic Energy Research Institute (KAERI) started tritiated waste treatment studies in 1983, when Wolsong unit1 commenced the commercial operation. Thus, KAERI has carried out some studies on tritiated waste treatment technologies. Kori nuclear power plant unit 1 has permanently shut down in 2017. Related to the decommissioning of nuclear facilities, KAERI performs tritiated liquid waste and groundwater treatment studies. So, we present various tritiated waste treatment methods including new adsorption and membrane technologies in this paper.

Keywords: Tritiated, Waste, Treatment, R&D

A Review on Requirements for Radioactive Waste Management During Decommissioning of Nuclear Facilities

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Korea will soon become one of the countries that will begin to dismantle nuclear facilities, starting with Kori Unit 1. It is expected that a large amount of radioactive waste will be generated during the decommissioning phase of the nuclear power plant. The waste generated will be different from the waste during operation phase in terms of quantity, type, and properties. Most of the main matters of the waste management plan during operation are managed by focusing on the final treatment. However, in order to treat a large amount of waste at the time of decommissioning, it is necessary to manage from the source to the final stage, and in the meantime, it also needs to properly establish management measures such as characterization, packaging, transportation, and containers.

In order to effectively manage large amounts of decommissioning waste, it is inevitable to develop a management plan from the generation to final container, transport, and disposal. To do this, it will be necessary to analyze the set of requirements for establishing a waste management plan. Therefore, this study intends to describe the requirements that will be useful in the method of tracking and managing nuclear plant decommissioning waste from its generation to final disposal.

First, the part to be considered in the process from generation of waste to final disposal is waste generation information such as origin, waste stream, mass, volume, radionuclides (dose rate and contamination), waste classification, processing, location, and so on. Origin provides information on equipment, system, area, etc. on the source of the waste. Processing represents the management process, and the location includes the generation location, current location, transport package and container ID, and so on.

Second, a plan for waste containers should be established. In operation phase of nuclear power plants, containers of 200 liter drums have been mainly used. However, in the case of decommissioning projects, alternatives for various types of decommissioning containers should be developed according the stream and classification of the waste. Information related to waste containers should include container ID, type, volume, mass, waste classification, dose rate, surface contamination, and so on.

Lastly, for systematic information flow and tracking related to waste management, a computerized management system will be an effective management tool. In this software, some aspects of contents mentioned above such as identifying waste items, containers, locations, waste stream, container types, waste types, waste classification, measurement methods, waste processing methods, disposition methods, etc. should be included and operated. The disposition method should covers plans for the interim storage, final disposal, and clearance, etc.

Above all, it will be of utmost importance that the requirements are properly reflected in the process and procedure to be established including the various items described above so that effective management can be achieved in stage of actual dismantling work. Eventually, the decommissioning waste management system should be able to control the inventory of generated waste and their containers. Identified waste items generated should be tracked and managed according to the established process, which should enable tracking of the entire process from the source information and area to the final stage of disposal. In this process, characterization and classification of waste needs to be defined, and it would be useful if implemented to provide documentation for waste packages and reports.

It is not easy to manage the waste from generation to the final stage. A complex work process could be constructed, and it would also have links with other decommissioning project management areas such as schedule, cost, safety, etc.

Keywords: Decommissioning, Waste management, Waste tracking, Project management system

All-in-one Radioactivity Measurement and Decontamination System With Secured Reliability

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As the number of nuclear energy utilization facilities in domestic increase, the amount of radioactive waste generated is increasing. In particular, a large amount of radioactive wastes is generated from nuclear power plants in operation and from the decommissioning of nuclear power plants in the near future. These wastes should be processed the self-disposal or the permanent disposal through measurement, classification, cutting, decontamination and other technical treatment.

We conduct a study to secure the reliability of measurement and decontamination. By automating the contamination level measurement and decontamination procedure is minimized the effect of the volume of radioactive waste. And it is configured to prevent human errors due to the measurement height and measurement speed, which vary depending on the worker measuring the contamination level, and to check the exact location of the contaminated area and decontaminate only the contaminated area to generate a related report. In addition, a suction device and an exhaust filter are separately designed to prevent the diffusion of contamination that can occur when drying decontamination through a laser, thereby preventing the diffusion of contamination in a workplace and preventing the internal exposure of the worker during decontamination.

Furthermore, the results of this study could contribute to increasing the self-disposal, reducing radioactive waste to be the permanent disposal, and minimizing worker radiation exposure by being used for measurement and decontamination of radioactive waste from nuclear energy utilization facilities.

Keywords : Radioactive waste, Decontamination, Laser, Contamination level, Measurement

Sensitivity Analysis of Groundwater Characteristics for Nuclear Power Plant Offsite After Decommissioning

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Individuals who present offsite of a nuclear power plant after decommissioning are subject to radiological effects by the diffusion of radionuclides on the site. The exposure pathway, which mainly has a radiological effects to the offsite, is groundwater release pathway. Therefore, this study conducted a sensitivity analysis of groundwater characteristics that should be considered primarily in assessing the radiological effect on the offsite individual.

Radiological effects to the offsite were evaluated based on RESRAD-OFFSITE. The RESRAD-OFFSITE code considers groundwater release and atmospheric diffusion of radionuclides to evaluate radiological effect on the offsite of contamination site. H-3 is easy to move through the water pathway in the form of tritiated water. Therefore H-3 was selected as a radionuclide for sensitivity analysis. The ratio of groundwater/surface water usage and the direction of groundwater flow were selected as input parameter for sensitivity analysis. In the case of groundwater flow direction, the sensitivity analysis was conducted by dividing it into 8 directions based on wells. Other input parameters used RESRAD's default values.

As a result of the sensitivity analysis according to the ratio of groundwater/surface water usage, 1.19×10^{-3} mSv/yr and 8.35×10^{-10} mSv/yr were evaluated when groundwater and surface water usage ratios were 1, respectively. As a result of the sensitivity analysis according to the direction of flow of groundwater, 1.19×10^{-3} mSv/yr was evaluated if the direction of flow of groundwater pass through the well of offsite. 1.16×10^{-3} mSv/yr was evaluated if the direction of groundwater flow is 45 degrees based on wells. If the direction of groundwater flow is 90 to 270 degrees based on wells, it was evaluated as 4.16×10^{-12} mSv/yr.

In this study, sensitivity analysis was performed according to groundwater characteristics for nuclear power plant offsite after decommissioning. In the case of groundwater/surface water usage ratio, the exposure dose decreased as the surface water usage increased. In the case of underground water flow direction, if the direction of groundwater flow is more than 90 degrees based on wells, the effects of groundwater are almost eliminated. Therefore, groundwater properties suitable for the offsite of nuclear power plant should be used for offsite dose assessments. The results of this study are expected to be available as the underlying data for offsite safety assessments.

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Keywords: Sensitivity analysis, Dose assessment, RESRAD-OFFSITE, Groundwater

Sensitivity Assessment of Exposure Dose for Geographic Information of Offsite After Decommissioning of Nuclear Facility

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Residual radionuclides exist on the decommissioning site of nuclear facility. These radionuclides are diffused around the site due to influences such as groundwater leaching, surface water release and air diffusion. Therefore, geographic information of offsite is expected to have one of the major effects on the exposure dose of individuals of offsite. This study performed the sensitivity assessment of exposure dose based on the geographic information of the offsite after decommissioning of nuclear facility.

For the sensitivity assessment, the RESRAD-OFFSITE computer code was used. C-14, Co-60, Ni-63, Sr-90, Cs-137 were selected based on foreign decommissioning cases. For geographic information, the candidate groups were selected using Geographic Information System (GIS). The geographic information of offsite includes dwelling site, agricultural land, pasture, and surface water body. Based on the selected candidate groups, the sensitivity assessment was performed by dividing the case where the surrounding sites were distributed and the case where they were concentrated. For other factors, the default values of the RESRAD-OFFSITE computer code were used.

As a result of assessment, it showed that in most of the radionuclides, the exposure dose decreased when the surrounding sites were distributed compared to the concentrated one. It appeared differently according to each radionuclides, and the exposure dose decreased up to about 70%. In the cases of Co-60 and Cs-137, the exposure dose due to external exposure decreased dominantly. Co-60 decreased by 36% and Cs-137 decreased by 41%. In addition, in the cases of Ni-63 and Sr-90, whose internal exposure by ingestion is the main exposure pathway, the exposure dose by ingestion of plant, meat, and milk decreased dominantly. Ni-63 decreased by 63% and Sr-90 decreased by 71%. In the case of C-14, the exposure dose was almost unchanged. Therefore, in order to derive reliable exposure dose, the geographic information of offsite should be applied appropriately. The result of this study can be used to evaluate the exposure dose of offsite after the nuclear facility is decommissioned.

Acknowledgements

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Keywords: Nuclear facility, Decommissioning, Offsite, Geographic information, Sensitivity assessment

Review of Foreign Cases for Dismantling of Nuclear Facilities

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The research reactor 1, 2 located in Gongneung-dong, Seoul, Korea's first nuclear reactors were permanently shutdown in 1995. Accordingly, the decommissioning project was initiated in 1997. The research reactors are currently at the stage for building dismantling and site release. However, research reactor decommissioning project was the first decommissioning project in Korea, with no experience in decommissioning nuclear facilities at the start of the project. And the dismantling of the nuclear facilities has not yet been conducted in Korea. Therefore, it is necessary to review foreign cases for the dismantling of nuclear facilities. In this study, review was conducted on the San Onofre 1 in US, Calder Hall nuclear power plant in UK, Tokai 1 in Japan as the representative nuclear facilities where building dismantling was carried out.

In the case of SONGS 1, Sphere Enclosure Building (SEB) surrounded and enclosed the Containment Sphere. When removing the roof of SEB, large excavators were placed on the roof to remove the roof of the SEB. After removing the concrete with excavators, steel structure was removed. In the case of Calder Hall, when dismantling of cooling tower, explosive dismantling was adopted as the most efficient dismantling method in terms of safety and cost. Explosive dismantling involves placing about 60% of the explosives around the cooling tower's shell and legs. Deformation, rotation, and collapse of the structure into the basin directly beneath the tower were considered. In the case of Tokai, dismantling of building is in progress. Turbine building and fuel storage building were completely dismantled. In the case of turbine building, turbine, generator, condenser piping, and feedwater systems in the turbine building were removed. And fuel storage building was completely dismantled because the fuel storage building was no longer used.

In this study, review of foreign cases for dismantling of nuclear facilities was conducted as a basic study for research reactor's building dismantling. San Onofre 1, Calder Hall, Tokai 1 where the dismantling of the building was carried out were reviewed as a representative nuclear facility. Each nuclear facility was dismantled in a manner appropriate to the building. Result of this study will be contributed to establish the methodology of building dismantling of nuclear facilities in Korea.

Keywords: Nuclear facility, Dismantling, Strategy, San Onofre 1, Calder Hall, Tokai 1

Preparation for the Development of the Configuration Management System in a Small Modular Reactor

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The purpose of the configuration management of nuclear facility is to ensure that the construction, operation, maintenance, and testing of the physical facility meets the design requirements specified in the design document. In other words, it ensures accurate information, consistent physical shape, and operating characteristics.

Configuration management generally performs change management, requirements management, information management, and interface management. Recently, tracking management and baseline management for change management are being conducted. Baseline management defines project milestone, or documents approved at a specific point in time, and is used as criteria when changes occur.

In the nuclear industry, change management was emphasized in order to maintain consistency with design requirements, physical shape, and configuration information from a safety point of view. Requirement management defined and documented design requirements in consideration of design, licensing, code and standards, and owner requirements to ensure the safety of nuclear facilities. Information management is to maintain the consistency of all information related to nuclear facilities throughout the life cycle. For information management of nuclear facilities, it is important to efficiently manage design-based and license-based information from the initial design stage. Interface management controls the interface between organizational management, in which participants are assigned a primary role, and change processes that occur throughout the life cycle of a nuclear facility.

After all, configuration management is to maintain design margin and operating margin in order to maintain the performance and operating conditions of nuclear facilities SSCs (structures, system, and components) derived from the design base and licensing requirements. In other words, this margin management provides reliability for safe operation of nuclear facilities.

The advantages of the developed small modular reactor project document management system are as follows. First, it has good processing speed, throughput and response speed by using a standard platform. Second, system security is excellent through ID, password, IP check, and security solution. Finally, convenience was improved by implementing a user-oriented system. After the development of this document management system, more than 3,000 documents were registered while operating normally for more than two years. It was decided to implement the configuration management function within one site, the document management system.

Keywords: Baseline management, Change management, Requirements management, Configuration management

Development of a New Method for Removing Si in Radioactive Concrete and Soil Wastes

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In Korea, 24 units of nuclear power plants (NPPs) are currently being operated and 2 NPP units, Kori Unit 1 and Wolseong Unit 1, have been permanently shut down as of 2021. A variety of radioactive wastes will be generated through decommissioning of Kori 1 and Wolseong 1. The generated wastes can be classified into metal, concrete, soil and other materials. Among them concrete and soil wastes are expected to account for more than 80%.

Concrete and soil mainly consists of silica. The high concentration of Si elements in these wastes has a large masking effect on the analysis of radionuclides. Also, it increases a background level of analytical equipment, negatively affecting the quantification of trace elements. Therefore, it is necessary to selectively remove the high Si contents to increase the sensitivity of measurements.

To find an effective condition for removing Si, the sea sand was dissolved in a mixture of strong acids. Subsequently, several drops of LiCl, NaCl, KCl, or RbF solutions were added to the solution. After centrifugation, the precipitate was separated from the solution and the supernatant was analyzed by Inductively-coupled Atomic Emission Spectrometer (ICP-AES) to determine the concentration of Si. The addition of RbF was the most efficient method to remove Si in the solution. In the stock solution treated with RbF, the Si content was lowered from 97.3 to 0.14% , improving the detection limit of Mo and Ti from 25 to 2 and 7.13 ppm, respectively.

In conclusion, the developed method in this study will help to analyze trace elements in concrete and soil wastes.

Keywords: Decommissioning, Radioactive wastes, Concrete, Removal of Si, Precipitation

Analysis on Major Cutting Techniques of Metal Waste Generated From the Decommissioning Process of Pressurized Heavy Water Reactor and Cutting Velocity

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On April 22, 1983, a total of four Pressurized Heavy Water Reactor (PHWR) were constructed in Korea, signaling the first commercial operation of nuclear plants with heavy water reactors. As for Wolsong 1, the government policy led to the decision to permanently shut down the plant and the change in the operation status to permanent shutdown has been approved, laying preparations for its decommissioning.

The major structures of the reactor systems of a PHWR include calandria, steam generator, pressurizer, and pipes and pumps of the heat transport system. The major metals used for these systems are ASME SA 105, 508, 516, Inconel 718, and 800, with each material having different components from on another.

A proper cutting method should be employed depending on the structure that is to be cut and the working conditions when dismantling nuclear facilities. There are three major cutting techniques, which have been analyzed in terms of their major characteristics: mechanical cutting, thermal cutting, and others.

This study chose the diamond-saw cutting, one of the mechanical techniques, out of the many cutting methods, and calculated the cutting velocity of pipes whose size ranged from 1 inch to 30 inches, which are major sizes for those used in a PHWR. The velocity is determined based on the blade diameter and the speed of the spindle, and the calculations indicated that it averaged at 75.5 inch/min, with the maximum speed amounting up to 127 inch/min. The range of the cutting velocity may vary depending on the conditions of the blade and the working environment of operators. It is expected to conduct a further study to estimate the cutting velocity per cutting technique based on the data on characteristics of each metal.

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Keywords: CANDU, Decommissioning, Metal Radioactive Waste, Cutting method, Cutting Velocity

A New Silicon Drift X-ray Detector Based ^{55}Fe and ^{59}Ni Spectrometer

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In this study, we developed silicon based x-ray detector for the quantitative measurement and analysis of ^{55}Fe and ^{59}Ni. These nuclides are currently analyzed through liquid scintillation counter or low energy germanium detector. However, it has several disadvantages; absence of auto sample feeder, regularly maintenance for new liquid nitrogen supply, and generation of radioactive organic waste by using liquid scintillation cocktail. To solve this problem, we utilized the silicon drift detector and named this system with 'Low Energy x-ray detector for advancing radIOactive waste characterizatioN' (LEXION). This detector is in operation at room temperature and able to identify decreased background radiation with low noise, simplified procedures, and higher resolution advantages compared to conventional analysis method. The energy resolution of LEXION shows FWHM is about 149 eV and the detection efficiency of ^{55}Fe and ^{59}Ni show 4.28 and 4.66 %, respectively. Also, the effect of ^{63}Ni beta ray induced characteristic x-ray, coexisting with ^{59}Ni, on the measurement of ^{59}Ni by LEXION was investigated.

Keywords: Decommission, Radioactive waste, Silicon drift detector, Low energy x-ray spectrometry

Decommissioning Radioactive Waste Classification System in Domestic and International

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Wastes generated during nuclear dismantling are classified into radioactive waste and non-radioactive waste. Radioactive waste is subdivided according to the specific activity of the waste and is classified in various ways according to the type. Internationally, radioactive waste classification systems and stabilization methods of nuclear power plants vary widely from country to country. In Korea, France, Spain, and United States, specific activity is used in classification, and the level of application differs from country to country.

In the United States, long-life and short-lived nuclides are classified as Classes A, B, and C as the classification limits. In Sweden, the surface dose rate is classified into very low level radioactive waste (<0.5 mSv/h), low level radioactive waste (<2 mSv/h) and intermediate level radioactive waste (>2 mSv/h). And in Canada, the classification system is classified into contact dose rate (≤2 mSv/h~>150 mSv/h) and working distance dose rate (<10 mSv/h~≥10 mSv/h).

The domestic radioactive waste classification standards are described in the 'Enforcement Decree of the Nuclear Safety Act', and were revised by introducing the IAEA classification standards in 2013. It is classified into high level waste (HLW), intermediate level waste (ILW), low level waste (LLW), very low level waste (VLLW) and clearance waste (CW). According to the domestic radioactive waste classification standards, all decommissioned waste, except spent fuel, are classified as intermediate level radioactive waste or less. Intermediate level waste corresponds to radioactive waste that exceeds the limit of radioactive waste, and the disposal methods is rock cavern. Low level waste is lower than the radioactive concentration limit of low level waste and corresponds to a waste that is more than 100 times the allowable concentration for clearance waste. Very low level waste is a waste that is less than 100 times the allowable concentration for clearance waste and higher than the clearance waste concentration.

Most of the dismantling waste from nuclear power plants is expected to generate low level waste and very low level waste. Clearance waste is general or industrial waste, not radioactive waste, free from application of the Nuclear Safety Act. They are substances managed by methods such as incineration, landfill, and recycling, and are expected to be generated in large amounts when dismantled. However, considering domestic public acceptance, it is considered difficult to treat it like general waste, and it is necessary to conduct research on ways to recycle it.

Keywords: Classification, Radioactive waste

Permanent Stationary Nuclear Power Plant Transitional Radiological Characteristics Evaluation Element Analysis to Ensure Sample Representativeness

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Kori Unit 1, which began commercial operation in 1978, was permanently shut down in June 2017, and is safely managed in a transitional state in present. If the decommissioning of nuclear power plants (NPPs) is detaily decided in near future, it is necessary to perform a radiological characteristics evaluation (RCE) of this NPPs in advance to use it as basic data for the decommissioning.

The RCE of transitional NPPs can be classified into six stages: HSA analysis, nuclide inventory evaluation, planning, measurement and sampling, analysis and evaluation, and validation. In order to secure the reliability of RCE results, the reliability of the procedures and results must be secured at every step. In general, The RCE of transitional NPPs mainly uses non-destructive methods because destructive methods cannot be applied. Currently available non-destructive measurement methods include radiation dose rate measurement, pollution level measurement using smear and detector, and gamma-ray spectroscopy that allows radionuclide analysis for various types of equipment and piping. Before securing the reliability of the measurement and analysis results, it is necessary to establish a standard for securing the representativeness of the sample and to determine the exact location and quantity based on this.

In this study, in order to secure the reliability of the RCE of transitional NPPs, the requirements for securing the representativeness of the sample were first reviewed, and main factors that directly influence the representativeness were derived. As factors for securing the representativeness of the sample, the object of the RCE, the measuring device, measurement location, the proper number of the sample, and the measurement procedure were investigated. In addition, the measurement methods required for the non-destructive radiological evaluation of the transitional NPPs and the statistical methods necessary to select the location and quantity of samples for measurement and analysis were investigated.

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Keywords: Transitional Nuclear Power Plant, Radiological Characteristics Evaluation, Representativeness of the Sample, Reliability

Review of Methodology and Cases of Radiological Impact Assessment for Transportation of Radioactive Waste of NPP

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Kori Unit 1, the first commercial nuclear power plant (NPP) in Korea, is under transition period of decommissioning, and the rest of the NPPs are expected to be decommissioned in future without further life extensions. Accordingly, containers are currently being developed for packaging, transportation, and disposal of various types, shapes, and sizes of wastes expected to be generated during decommissioning of the NPPs. Therefore, a preliminary risk assessment will be conducted for radiational impact during the transport of such radioactive wastes from each NPP site to the low and intermediate level radioactive waste disposal facility in Gyeongju. In this study, the assessment methodology and related cases are reviewed prior to preliminary assessment of radiational impacts from radioactive waste transport.

The radiological risk assessment is being performed by calculating the expected exposure doses of the general public and workers in the process of transporting radioactive waste and comparing it with the exposure dose limits. The required inputs for the evaluation include physical/radiological characteristics of wastes, specifications of transport containers, transport quantity, modes of transport (land, sea, etc.), population adjacent to the transport route, and transport scenarios. Additionally, accident conditions require information such as accident incidence rate and accident type (crash, fire, rollover, etc.). The individual/collective exposure dose for the general public and workers are calculated using RADTRAN, INTERTRAN, and RISKIND which is the risk assessment computer code. The RADTRAN is widely used for assessing radiational risk associated with transportation of radioactive waste by various modes, such as truck, rail, air, ship, and barge under routine and accident conditions.

As domestic evaluation cases, Korea Radioactive waste Agency (KORAD) and Korea Atomic Energy Research Institute (KAERI) each carried out radiological impact assessment. The KORAD conducted assessments for railroad transportation of high-level radioactive waste, road transportation of radioisotope waste, and road transportation of waste. KAERI evaluated an assessment on transportation of wastes generated during operating and decommissioning of research reactor. These organizations evaluated the radiation exposure dose to workers and the general public using RADTRAN code and compared the results with the domestic radiation exposure legal limits. Furthermore, the sensitivity of the expected exposure dose according to the change in the distance between workers and the waste and in the leakage rate of radionuclides in the waste packaging was evaluated. As per the evaluation, it was confirmed that the exposure dose in both normal and accident condition were less than the domestic annual exposure dose limits.

Keywords: Decommissioning Radioactive Waste, RADTRAN, Transportation Risk Assessment, LILW Disposal facility, Annual dose limit.

Hot Isostatic Press Analysis for Cement Solidification of Sludge Wastes From Decommissioning of the Nuclear Power Plant

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The optimum solidification conditions of sludge wastes by using on HIP (Hot Isostatic Press) were analyzed for the reduction of hardening time and the satisfaction of disposal criteria. The hardening step of cement radioactive waste solidification, which is made of sludge fixed in a radioactive contamination tank, is implemented in the environment of high temperature and pressure. For existing sludge type wastes, absorbents are injected to reach a suitable compositional range for cement solidification, and at least 10 wt% cement and 28 days hardening step are required to satisfy solidification criteria including compressive strength, leaching, etc. The composition of the cement solidification surface such as calcium hydroxide and calcium silicate hydrate are analyzed by XRD(X-ray Diffraction) due to changes in temperature and pressure conditions of hardening step.

The sludge targeted in this study was from stainless steel corrosion contamination tanks and are generated from liquids, gases, solid waste disposal systems and steam supply systems. The weight percent of sludge in stainless steel corrosion products consists of Fe_3O_4(59.5 wt%), Fe_2O_3(15.3 wt%), and Cr_2O_3(25.1 wt%). The composition of the sludge waste generated by the radioactive contaminant tank was derived through literature investigation research and a cement solidification was produced based on this. Cement solidification made from sludge containing Co-60 were estimated based on the KORAD (Korea Radioactive Waste Agency) criteria. Solidification evaluation methods were performed using the ANS16.1 for leaching evaluation, solidification surface compositions analyzed through XRD, and whether it could withstand the disposal compressive strength, 3.44 MPa.

Keywords: Hot Isostatic Press (HIP), Cement, Solidification, Sludge

Application of Hydrophobic Silica Nanoparticles for the Selective Separation of Fine Particles From Cesium-Contaminated Soil

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For the contaminated soil in Fukushima, it has been reported that the <5 μm size fraction accounts for approximately 7% of the topsoil (0–10 cm); the cesium distribution is higher in this fraction than in silt and sand, which make up the majority of the rest of the topsoil. For this reason, the removal of fine particles could lower radioactivity of the remaining soil. Previous results reported that radioactivity was reduced by more than 80% after the separation of suspended clay from a mixed soil suspension. Therefore, the separation of clay containing a high concentration of cesium from soil can be an effective strategy for radioactive soil decontamination. Several chemical/physical methods have been reported for the separation of cesium contaminated clay.

A new technology has been evaluated the feasibility for selective separation of fine particles in the cesium-contaminated soils using hydrophobic silica nanoparticles as surface modifier. The influence of the silica type and dosage were systematically examined for separation efficiency and interaction between silica nanoparticles and fine particles. The results showed that fine particles attached with hydrophobic silica nanoparticles can be separated in cesium-contaminated soils, forming the silica on the surface of fine particles due to the van der Waals attraction in their collisions. Column flotation tests were also conducted to evaluate the effectiveness of separation using hydrophobic silica nanoparticles. The results indicated that hydrophobic silica nanoparticles increased the attainable separation by 5 times and the maximum attainable separation from 20% without HPOS NPs up to 99.5% with 0.04 g- hydrophobic silica nanoparticles per g-soil. The cesium reduction in fine particle, which accounted for 53.2% of the total soil, corresponded to a high decontamination efficiency of approximately 58.3%. In conclusion, the attachment of hydrophobic silica nanoparticles on fine particles in the flotation process presents a significantly improved fine particle separation in cesium-contaminated soils, becoming an alternative separating fine particles without surfactant to facilitate the secondary waste disposal.

Keywords: Cesium, Fine-particles, Hydrophobic-silica-nanoparticle, Flotation, Separation

Uranium Leaching Characteristics From Soil Under Acidic/alkaline Conditions

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Uranium-238 (U-238) as a fertile material may transport the subsurface environment, and it produces radioactive contaminants (e.g., soil and groundwater) under the operation or decommissioning process of related facilities. The soil having high contents of uranium would be decontaminated to reduce their waste volume and permanent disposal costs. Washing treatment is commonly used to remediate/decontaminate the heavy metals/radionuclides from the soil. This study adopted soil washing method in order to treat the large volume of uranium-contaminated soil waste. The pHs of washing solutions were controlled with H_2SO_4 and $HaHCO_3$ to trigger the solubility of uranium ions. The leaching characteristics of uranium from the soil were compared under the different leaching conditions. This study cautiously suggests the optimized conditions for uranium decontamination from the soil for self-disposal treatment of the soil wastes.

Keywords: U-238, Uranium contaminated soil, Soil washing, Uranium leaching, Soil decontamination, Self-disposal

Review on I-129 Scaling Factor on Low-Level Waste

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In the field of radioactive waste management, Iodine-129 is quite important, due to their special characteristics. Since they have long half-lives, long-term safety of radioactive waste with the radionuclides should be assured. Also, they can exist as anion chemical form and this enhances their mobility and solubility in underground water.

Regarding disposal of low-level waste, radioactivity concentration of the waste can be analyzed using indirect measurement. Scaling Factor is a typical method of indirect measurement that can reduce the individual analysis and analysis efficiency may be improved by applying it to the mass waste such as decommissioning waste. It is evaluated from ratio of difficult-to-measure (DTM) nuclides, including I-129 and Key (KEY) nuclides which emit high energy gamma-ray such as Co-60 and Cs-137. However, Scaling Factor is easily overestimated, because radioactivity of DTM nuclides are usually based on detection limit. With conventional method of radiochemical analysis, such as gamma-spectrometry and gas proportional counter, detection limit is not low enough to detect true radioactivity of samples. Consequently, the detection limit value is reported instead of true value of radioactivity. Obviously, detection limit is higher than true value of DTM, and Scaling Factor based on it may be also higher than true ratio of DTM and KEY.

In this review of previous studies from relevant institutes, it has been shown that mass spectrometric method applied for analysis of I-129 can solve this issue. Sufficiently lower detection limit than conventional methods can be obtained from it. Thermal ionization mass spectrometry (TIMS) and inductively coupled plasma-mass spectrometry (ICP-MS) with special sample introduction was researched for obtain lower detection limit. Comparison of conventional method and proposed method on low-level waste is presented in this review. The mass spectrometric method and relevant Scaling Factor can be applied for large amount of low-level waste such as decommissioning waste.

Keywords: Radioactive Waste, Low-Level Waste, Scaling Factor, I-129, Decommissioning Waste

Overview of Data Quality Objective in Characterization Survey

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Characterization survey, a major step for site release after decommissioning of nuclear facilities, is performed to estimate the extent of surface and soil of the facilities exceeding the release criteria. This survey checks the status of decontamination for the surface and soil, determines treatment and disposal options of radioactive materials, and generates the most comprehensive information. Because insufficient characterization information due to incorrect assumptions results in unexpected time and cost, an iterative review process is required and the process can be performed based on DQO (Data Quality Objectives).

DQO, developed by US Environmental Protection Agency, collects the data needed for decision making, ensuring that the right quality and amount of data. DQO helps guide the process for formulating a problem, identifying the decisions, specifying the quality requirements for the decisions that lead to the quality requirements for the data and finally developing a defensible sampling and analysis plan. This structured planning process can help lead to the collection of "good" environmental data to support decision-making especially in characterization survey. There are seven steps in DQO as below.

Step 1 is to clearly define the problem.

Step 2 is to define the decision that will be resolved using data to address the problem.

Step 3 is to identify the informational inputs to resolve the decision.

Step 4 is to specify the spatial and temporal circumstances that are covered by the decision.

Step 5 is to integrate the outputs from previous steps into a single statement.

Step 6 is to specify the decision maker's acceptable limits on decision errors.

Step 7 is to identify the most resource-effective sampling and analysis design for generating data.

In the design of characteristic assessments, specifying two-step decision requirements and specifying five-step decision rules are critical processes, and characteristic assessments should be carried out gradually through a step-by-step approach and iterative process in case more conditions change than expected.

Data quality goal-based characteristic evaluation conducted at the actual site is diverse and complex sampling strategies are essential. It is necessary to reduce the uncertainty of the plan using the appropriate and valid methods when they exist. From this point of view, the data quality objective process reduces potential risk and provides the right quality and appropriate amount of data by performing an iterative process for specific decisions, so appropriate use in the decommissioning characteristic assessment process will help efficient and effective project execution.

Keywords: Data Quality Process, DQO, Characterization, Decommissioning

Overview of Procedure to Release the Survey Unit in MARSSIM

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In order to release the site after decommissioning the nuclear power plant, the concentration of radioactivity on the site shall not exceed the criteria for release of the site and remaining buildings after the decommissioning of the nuclear facilities is completed. United States is verifying that the criteria are met through the MARSSIM (Multi-site Radiology and Field Surveys Manual) procedure jointly developed by national agencies related to dismantling nuclear facilities. MARSSIM proposes a Sign test method to verify that the residual radioactivity concentration at the site meets the Derived Concentration Level (DCGL) corresponding to the limit of the residual radioactivity concentration applied for site release.

The purpose of this study was to determine whether the contamination level at the site met the residual radioactivity induction criteria for unlimited reuse (0.19 m/y) at the time of the Connecticut Yankee decommissioning project, and to analyze the application of the nonparametric methodology.

At the time of decommissioning the Connecticut Yankee nuclear power plant (CYNP), the concentration of residual radioactivity on the site was determined and compared to the residual radioactivity standard presented in the US NRC Report (LTP). CYPN conducted a Sign test, a nonparametric statistical method proposed by MARSSIM, because no major radionuclides were found in the field background. There are some functions in Sin test, which are percentiles determined by the selected decision error levels α and β which are probabilities of incorrectly releasing contaminated survey units and failure to release unpolluted units, respectively. It also has a probability that random measurements within the survey unit are less than DCGL. The interval between samples can be obtained by utilizing area and number of samples.

Performing the Sign test on a survey unit in CYNP, the required number of samples was obtained by applying factors reflecting the characteristics of the power plant, and the number of times that the difference between DCGL and the weighted sum of concentrations for each sample was positive. By comparing the weighted sum with the critical value, threshold value corresponding to the number of samples with of the decision error, it was confirmed that all sample's concentrations were smaller compared to the DCGL. Accordingly, the survey unit satisfies the site release criteria, and it could be released as an unrestricted reusable site.

Keywords: MARSSIM, Decommissioning, Nonparametric methodology

Preliminary Study of the Optimal Scale Up Factor for the Pilot Scale Chemical Decontamination Liquid Waste Decomposition System

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In prior to develop commercial scale chemical decontamination liquid waste decomposition system (CWDS), KEPCO KPS performed pilot scale test applying advanced oxidation process using two types of UV lamps (low and medium pressure) under various concentration of iron and hydrogen peroxide. Based on the test result, electric energy per order was calculated, one of two scale up factors, in every condition and feasibility to the pilot scale CWDS was evaluated.

In the experiment, pilot scale advanced oxidation process system, which is consisted of low and medium pressure UV lamp (capacity in kW), heat exchanger and the loop test system. Test was conducted by irradiating UV light to process water with continuous flow until decomposition rate achieved 95%. To simulate the condition of process water in nuclear power plant, certain concentration of iron chloride was added. For UV advanced oxidation process, hydrogen peroxide was also added. Determination of degree of decomposition was performed by the calculating scale-up factor, electric energy per order (E_{EO}).

The result of the experiment showed that process water with iron chloride with 22 mM of hydrogen peroxide, under both low and medium pressure UV irradiation, has lower E_{EO} value. This indicates that lower UV energy is necessary to achieve 95% of decomposition.

UV advanced oxidation process was applied to chemical decomposition system and electric energy per order was calculated as a factor to determine its feasibility to pilot scale test. Each test condition showed patterned result as the concentration of iron and hydrogen peroxide varied. However, these values were significantly different even with the same test condition. Therefore, we concluded that electric energy per order was not appropriate as a factor to determination of the feasibility of pilot scale advanced oxidation process.

In further research, additional experiments will be performed at the same condition. However, to determine the scale up factor, electric energy per mass which is known as another scale up factor and more applicable when the species being destroyed at a high concentration (>100 mg/L) will be applied rather than energy per order.

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Keywords: Chemical Waste Decomposition, UV AOP process

Preliminary Studies of Conditional Clearance of Concrete Waste in Radioactive Waste Repository Using AMBER Software

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A number of nuclear power plants (NPPs) around the world will be shut down and accordingly demands of decommissioning NPPs will be increased. During decommissioning of NPPs, a lot of radioactive concrete waste were generated and most of them were clearance waste or very low-level waste (VLLW). In the past, clearance waste was disposed by landfill, incineration or recycling and VLLW was disposed as radioactive waste to the disposal facility. Clearance waste was recycled in public industries but there had some problems with public acceptance due to radiation exposure. Therefore, restricted recycling which was recycled in nuclear sector is considered as alternative way which affect less radiation exposure to public. If radioactive concrete waste was recycled in nuclear sector, a higher level of radioactive waste will be recycled because a target of exposure is radiation workers. In EUR-18041 report, scenarios of recycling radioactive concrete waste in nuclear sector as disposal products using in a repository were suggested. It needs to be considered that radiological assessment after disposing recycled products in the repository. AMBER software was used in the safety assessment of a repository.

In this study scenarios of recycling as backfill materials or disposal container in vault repository was considered and post-closure safety assessment of a near-surface disposal facility which filled with backfill materials or vault cubes was conducted. ISAM vault test case which was modeling for the safety assessment of a near-surface disposal facility due to liquid leakage was suggested in AMBER software. Post-closure safety assessment of the vault disposal facility was conducted based on ISAM vault test case. Compartments of the repository were consisted of waste and concrete base compartments in ISAM vault test case. Therefore, post-closure safety assessment was conducted by subdividing compartment as vaults, backfill materials, vault cubes and waste. Addition of radioactivity inventory in backfill materials or vault cubes due to using recycled radioactive concrete was calculated by multiplying radioactivity concentration applied either of clearance level or VLLW level and mass of concrete required in compartments. Finally, the effect of radioactivity addition due to recycling for vault cube or backfill materials for annual public dose was analyzed.

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Keywords: Restricted Recycling, Decommissioning, Radioactive Concrete Waste, Clearance, Safety Assessment

Big Data Analysis With R-studio for Radioactive Waste's Concentration Behavior According to Data Components

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In the nuclear industry, decommissioning of nuclear power plant is becoming one of important issues. So, Information or prediction on radioactive waste that generated during decommissioning is required. In this paper, big data analysis with R code and R-STUDIO for radioactive waste concentration behavior that calculated with SCALE 6.2's ORIGEN-ARP code is conducted. In the ORIGEN calculation, 4 materials are chosen as nuclear power plant radioactive wastes such as concrete, Inconel-718, carbon steel, stain-less steel 304. And many calculation components are considered such as neutron flux, material impurity, cooling time, etc. In the Inconel-718 impurity calculation, impurity is not considered because of no impurity reference. ORIGEN-ARP calculation result is processed as specific activity term such as Becquerel / gram unit. In the dataset, over 120,000 isotope's specific activities are calculated. For simple analysis, specific isotope trends are chosen such as Cs-137 and Co-60. To make it easier to see a trend of radioactive waste concentration, concentration-flux plot figures are made with GGplot2 that application in R-studio and result dataset. As a result, usually, the higher the flux, the higher the calculation result is gotten. And most of result, Inconel-718 has highest results and less distributed results. Unlikely Inconel-718, other materials have evenly distributed results. In the same flux for Co-60, Inconel-718 waste's specific activity is higher than any other material, concrete waste's specific activity is lowest. In the Cs-137 analysis, similar trend is appearing with Co-60, but specific activity is much lower than Co-60's. With these results, it is expected to be useful in decommissioning nuclear power plants.

Keywords: Big Data, Radioactive waste, R-studio, Origen-ARP

Sensitivity Study of Cobalt Self-Powered Neutron Detector

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Self-powered neutron detectors (SPNDs) are the instruments which are belongs to in-core devices to detect the neutrons in the reactor. The existing in-core detectors are mostly made of Rhodium and Vanadium. In this study, the sensitivity calculation has been performed with several conditions of the cobalt SPND. The modeling and sensitivity calculation has been performed with MCNP6.2 code, and the modeling of the cobalt SPND has been visualized with Visual Editor.

For the sensitivity study, the radius of the emitter, insulator and the collector has been changed with ±0.005 cm intervals. In addition, for detailed sensitivity calculation, two spectrums have been proposed as neutron source, one is Watt fission spectrum, and the other is the neutron spectrum which calculated from the CE 16x16 type fuel assembly. The results are compared between the change of emitter, insulator, and collector radius, and compared between two spectrums which proposed in this study with optimized model of cobalt SPND.

The sensitivity calculation of this study will be one of the basic data for the improved cobalt SPND design.

Keywords: Cobalt, SPND, Sensitivity, MCNP

Removal of Uranium Ions From Soil Washing Effluent by Neutralization Treatment

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The Korea Atomic Energy Research Institute has been conducting research on decontamination of uranium-contaminated soil through the soil washing method. Uranium could be separated from the soil by acidic or alkaline soil washing agents such as H_2SO_4, $NaHCO_3$, and Na_2CO_3 which increases the solubility of uranium. In order to recycle and final treatment of the soil washing effluent, uranium ions must be removed from the effluent by treatment methods such as precipitation, adsorption, and ion exchange. In this study, precipitation method was selected to separate uranium ions from the effluent using neutralization treatment. After the neutralization treatment, the uranium concentration of supernatant was analyzed using X-ray fluorescence spectrometer, and the properties of the precipitate were investigated by SEM-EDS analysis. Also, we compared the amount of precipitate generated during neutralization treatment of acidic or alkaline effluent.

Keywords: Uranium decontamination, Soil washing, Neutralization, Precipitation

Comparison Between Decision-Making Approaches for Evaluating Decommissioning Waste Treatment Process

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In this study, a comparison between decision-making approaches was performed to select the appropriate technique according to the selection criteria for decontamination. AHP, PROMETHEE, and TOPSIS were selected as approaches for comparison.

Since decommissioning is a project that requires enormous cost and manpower, studies have been conducted to select the optimal process. However, the existing methodology had two limitations: (1) They could not reflect changes according to the target because preference was derived using a single criterion. (2) It cannot reflect the situational conditions such as the business operator's target budget and the amount of waste.

In order to select appropriate criteria for decontamination, a survey was conducted with subject-matter experts. As a result, it was confirmed that safety is the top priority for LLW, but waste minimization is the most important consideration for VLLW. This result shows that, as the radioactivity level of the target decrease, it is more important to consider that there is no additional waste due to decontamination when selecting a process. Also, it confirms the assumption that different criteria should be considered depending on the characteristics of the object.

Using the derived preferences, a comparison between decontamination techniques was performed targeting the steam generator of a nuclear power plant. Quick decision-making is possible by utilizing numerical comparisons between scenarios. It is expected that it will be able to help in establishing a decommissioning plan for domestic nuclear facilities that are entering the decommissioning stage starting with Kori Unit 1.

This study was investigated by Korean experts, and the results may be different in other countries because the method and cost of treating radioactive waste are different from those in Korea. Comparing them could also be meaningful future studies. In addition, it was difficult to quantify the performance of decontamination techniques because there are no standardized experimental conditions. If standardized experimental conditions are proposed, it will be easier to perform scenario construction.

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Keywords: Waste treatment, Decommissioning, Decision making process, AHP

Development Trend of Thermal Treatment Technology

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In the decommissioning step of nuclear power plant, various radioactive wastes are generated. Unlike general industrial waste, radioactive waste requires suitable treatment technology to satisfy the waste acceptance criteria of the repository. Therefore, many countries are developing waste treatment technologies to make radioactive waste disposable.

There are various types of technologies for treating radioactive waste depending on the target waste. Among them, thermal treatment technology is capable of disposing of various types of waste (organic liquids, inorganic liquids, organic solids, inorganic solids, mixed organic-inorganic solids, mixed organic-inorganic liquids, Spent resins). KHNP also has been developing CCIM (Cold Crucible Induction Melter) and PTM (Plasma Torch Melter) for thermal treatment of waste.

And now, in the European Union, an evaluation of the developed thermal treatment technology is underway. Thermal treatment technologies are under evaluation include CEA’s SHIVA and In-Can melting, NNL’s GeoMelt and HIP (Hot Isostatic Pressing), VTT’s Thermal Gasfication and VUJE’s VICHER. For the practical commercialization of these technologies, the target waste evaluation, economic feasibility evaluation, operation safety, environment impacts and disposal suitability evaluation are being performed for the target wastes.

KHNP is also planning to conduct an evaluation on the applicability of thermal treatment technology and disposal suitability of radwaste.

Keywords: Decommissioning, Radwaste, Thermal technology, Disposal

Review on High Volume Reduction Technologies to Ensure Decommissioned Radioactive Waste Treatment and Disposal Suitability

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When looking at the generation of waste during the decommissioning of nuclear power plants in Korea, it was estimated that about 92 vol.% of industrial waste and about 8 vol.% of radioactive waste contaminated with radioactive materials. The types of radioactive waste generated during decommissioning are metal, concrete, soil, sludge, DAW, asbestos, mixed waste, etc. A large amount of radioactive waste is expected to be clearance waste through the decontamination process. However, for radioactive waste still to be sent to the repository, it is expected to exceed the national goal of 14,500 drums (200 L basis) per nuclear unit. Therefore, it is necessary to reduce the decommissioned radioactive waste by super volume. In this review, several techniques that can be used in a disposable form at the Gyeongju disposal site were examined. Various decommissioned radioactive wastes cannot be disposed of with single technology. As a technology that can reduce a large amount of metal, the application of induction melter is the most promising. If this technology is applied, the volume reduction factor is about 5. The super compaction technology, which compresses with a force of 2,000 tons, is applicable when the radioactive waste itself can be disposed of. When this technology is applied, the volume reduction factor is about 4. Radioactive wastes that cannot be disposed of without changing their physico-chemical properties must be thermally treated, and the most powerful technology is the plasma torch melting technology. When this technology is applied, the volume reduction factor is about 5. In this review, appropriate treatment and reduction techniques for each type of radioactive waste were reviewed, and reduction effects were also evaluated.

Keywords: Decommissioning, Radioactive waste, Induction melter, Super compaction technology, Plasma torch melting technology

Wastewater Treatment by Chemical Precipitation and Adsorption After Soil Washing From Cesium Contaminated Soil

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A large amount of washing wastewater is generated after decontamination of cesium contaminated soil using acid treatment. Radioactive liquid waste contaminated with ^{137}cesium radionuclides must be treated effectively due to its long half-life, high solubility/mobility, and high radiotoxicity. Various chemical, physical, biological and combined methods have been developed to decontaminate radioactive wastewater. Among them, physicochemical methods include adsorption and ion exchange, advanced oxidation process, and chemical precipitation. In addition, electrochemical methods have been studied, such as electrodialysis, electro-adsorption and electro deionization to decontaminate radioactive wastewater. Additionally, membrane has shown great potential for decontamination of radioactive wastewater due to its unique pore-sized-dependent separation mechanism. The methods include reverse osmosis, nanofiltrtion, membrane distillation, and forward osmosis. Among the various methods to treat radioactive liquid waste, chemical precipitation is an effective and the most widely used process in industry, because it is relatively simple and inexpensive to conduct. In the precipitation process, our previous study conducted that chemicals ($Ca(OH)_2$) react with metal ions to form insoluble precipitates, and cesium also reacts with precipitants (NaTPB) to form insoluble salt precipitates. The forming precipitates can be separated from the water by sedimentation or filtration which resulted in the reduction of radioactive waste volume.

In this study, we measured the efficiency of the cesium desorption from a sand (0.2 mm>), using 0.5M nitric acid. In the wastewater treatment process, we purified the wastewater through a two-step process to remove metal ions and cesium ions. In the first step, hydroxide precipitation using $Ca(OH)_2$ was treated to remove metal ions, and in the second step, cesium ions were removed by treatment with various adsorbents. We found the optimal dosage to remove metal ions and cesium from nitric acid washing wastewater and investigated the optimum pH range of the solution, and we confirmed that the ceium and metal ions were significantly reduced by the two-step treatment.

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Keywords: Cesium, Soil, Wastewater, Treatment, Precipitation, Adsorption

A Case Study on Decontamination Method and Soil Characteristics at Overseas

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The first commercial reactor, Kori Unit 1, was permanently shut down in 2017 and is set to be decommissioned. As it is the first attempt to dismantle in Korea, unexpected radioactive contamination may occur during decontamination and dismantling process, and the large gap between the expected amount and the actual amount of radioactive waste leads to an increase of decommission costs. Therefore, recognizing the effective decontamination process and effectiveness in antecedents' case of overseas is an important challenge. Therefore, in this study the decontamination process will identified, used in the three nuclear power plants in the United States, Connecticut, Maine Yankee, and Yankee Rowe, which were dismantled immediately, the final site condition such as Kori Unit 1.

In addition, since the site opening condition for Kori Unit 1 is green remediation, the concentration and DCGL will be compared by radionuclide of the soil to confirm the criteria for clearance. Through this, effective decontamination technology will be recognized and significant points will be presented when decontamination and dismantlement of Kori Unit 1.

Keywords: Kori Unit 1, D&D, Decontamination technology, Radioactive waste, Soil characteristics

PTC Analysis Method for Evaluating the Diffusion Depth of Radionuclide

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It must be accurately identified that radionuclides, radioactivity levels, and diffuse contamination through radiographic characterization of soil or concrete structures when decommissioning nuclear facility. This is because the amount of radioactive waste generated from soil or concrete structures can be predicted and reduced. In this study, the validity of the PTC (Peak to Compton Count Ratio) method was evaluated to evaluate the contamination diffused to the depth based on the gamma spectrum acquired obtained from in-situ measurement.

The gamma spectrum was measured for a total depth of 20 cm by moving the target at 2 cm intervals from the surface for the representative gamma radionuclide for soil contamination (^{137}Cs) and the representative gamma radionuclide for radioactive concrete (^{60}Co, ^{152}Eu). The MCNP simulation is performed under the same conditions as the experiment and the Peak to Compton counting rate ratio (Q-value) is calculated according to depth. An optimum correlation Compton continuum area is established based on the Q-value calculated in the experiment and MCNP simulation.

Compared to MCNP simulation, the mean relative error was 2.93% (^{137}Cs), 1.66% (^{60}Co) and 2.45% (^{152}Eu) for a single source. 1.62% (^{60}Co) and 5.63% (^{152}Eu) for mixed sources. The correlation equation was derived by calculating the Peak to Compton counting rate ratio for depth. Based on the derived correlation results, it was confirmed that the high reliability for all depth. As a result of evaluating the depth by back substitution of the Q-value calculated from the Q-B correlation, the mean relative error was less than 9.0% (^{137}Cs, ^{152}Eu) and 4.0% (^{60}Co) a single source. And the average relative errors for mixed source were 12.0% (^{137}Cs, ^{152}Eu) and within 4.0% (^{60}Co).

It was confirmed that ^{60}Co has the highest reliability for both single source and mixed source. This is because natural radioactive nuclide do not affect the Compton continuum area produced by the high emitted energy of ^{60}Co. On the other hand, ^{152}Eu has a higher relative error than other nuclides because it has a lot of emitted energy and is affected by natural radioactive nuclides. However, considering that it is the result of in-situ analysis, it can be seen that the PTC method is valid for evaluating the contamination that has diffused into the interior. Based on these results, the PTC method is thought to can be useful for assessing radionuclide contamination that have diffusion depth in various structures, such as soil or concrete.

Acknowledgments

This is a study conducted as part of the Korea Atomic Energy Research Institute's (KAERI) ‘Development of the Advanced Decommissioning, Decontamination, and Remediation Technology for Nuclear Facilities’ collaboration project.

Keyword: In-situ, Gamma spectrum, Peak to Compton, Depth, Q-value

The Treatment of $^{14}CO_2$ Using Glass-based Adsorbent

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In the nuclear facility, gaseous radioactive materials ^{14}C, ^{13}H, ^{131}I are treated by activated carbon filter in the air purification system (Heat, Ventilation, & Air Conditioning) for environmental protection and safety of radiation workers. The spent activated carbon is regularly replaced because of radioactive materials. The spent activated carbon can be treated by thermos-chemically method for reuse. However, radioactive carbon is generated in the form of $^{14}CO_2$ during thermos-chemical process. The $^{14}CO_2$ should be removed at room temperature under atmospheric pressure to prevent re-release of $^{14}CO_2$ and to consider the stability of radioactive carbon. In this presentation, Sr^{2+} and Ca^{2+} which are alkaline earth metal ions are added to the glass structure and reacted with carbon dioxide in aqueous phase to mineralize CO_2 into carbonate form. When Sr^{2+} and Ca^{2+} ions are released from adsorbents, alkaline earth metal ions and HCO_3^- react to adsorb CO_2 in the form of carbonate. The reacted surface of the glass and CO_2 capacity were characterized by using XRD (X-ray diffraction), SEM (Scanning electron microscopy), TEM (Transmission electron microscopy), TGA-MS (Thermogravimetric Analysis/Mass Spectrometry), respectively.

Keywords: Adsorbent, CO_2 capture, Carbon-14, Glass

In-situ Radioactivity Analysis Using ISOCS for the High-level Radiation Source

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In-situ radioactivity analysis is a direct radioactivity measurement and analysis method that can directly measures an object to be analyzed and design a measuring geometry to perform radioactivity analysis immediately on site. And, it is widely used in the fields of decommissioning nuclear facilities, such as radiological characterization, prediction of radionuclide inventory, site remediation, etc.

In this study, to find out the accuracy of in-situ radioactivity analysis using ISOCS (In-Situ Objective Counting System), which is CANBERRA's representative in-situ radioactivity analysis system, in-situ radioactivity analysis was performed for a calibration source. In particular, the high-level calibration source was used as the object to be analyzed in consideration of high-level radiation components and sources such as the reactor coolant system (RCS) in nuclear power plants (NPPs).

The nuclide of the calibration source (point source) was ^{137}Cs, and the radioactivity was 1.54×10^9 Bq. ISOCS collimators (30°, 90°, 180°) were used, and Measurements were performed three times for each 25 to 70 cm section at measuring distance of 80 to 440 cm. As a result, in the case of a 30° collimator, it was analyzed that the uncertainty is 1.22 % at the measuring section where the dead time is less than 20%. also, the uncertainty has increased under certain test conditions (collimators and measuring distance) where the pulse pile-up and the dead time increase.

This study confirmed that the accuracy of in-situ radioactivity analysis using ISOCS for the high-level radiation source can be improved by selecting appropriate measuring conditions (collimators and measuring distances) that can reduce the dead time to less than 20% and the pulse pile-up. In the future, it will perform a test to improve the accuracy of in-situ radioactivity analysis using ISOCS for the certain radiation source in high-level mixed radiation fields such as the RCS in NPPs rather than a single radiation source.

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Keywords: In-situ, Radioactivity analysis, ISOCS, Dead time, Decommissioning

Synthesis of Pollucite Using Korean Natural Zeolite at Temperature Below 100°C

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Radioactive Cs-137 is one of the main issues in terms of both nuclear accidents and radioactive wastes. The monovalent Cs-137 is usually highly soluble in aqueous media, and it can be introduced into the human body muscles through ion-exchange to K^+. Therefore, many researchers tried to immobilize the Cs^+ into the final mineral phase or solid waste form.

Recently, pollucite ($CsAlSi_2O_6$) has become a promising material for the final disposal of Cs-137, because the Cs-137 can be trapped in the aluminosilicate structure of pollucite mineral. However, all of the currently developed methods using hydrothermal synthesis require high-temperature conditions around 200ºC. Even some researchers suggest that the threshold temperature for the synthesis of pollucite is 160ºC, it still needs to use stainless steel autoclave (Parr reactor).

Here, using Korean natural zeolite, we developed a synthesis method for the pollucite at 96ºC without using stainless steel autoclave. Korean natural zeolite consists of mordenite and clinoptilolite, and both zeolites have high Cs^+ sorption affinity. This experiment was initiated, assuming that sorption affinity can lower the threshold synthesis temperature required for the phase transition to pollucite. In our experiments, we first optimized the condition (natural zeolite: 4 g, Cs_2CO_3 2 g, NaOH 3 g, D.I. water 25 mL) of pollucite synthesis at 96ºC.

For the final disposal, it is necessary to verify that pollucite synthesized at 96ºC has a similar Cs immobilization performance as conventional pollucite. Thus, characterization was mainly focused on determining whether or not the pollucite synthesized at 96ºC (POL96) is different from those synthesized at 150ºC (POL150) and 200ºC (POL200). In X-ray diffraction and fluorescence analysis, POL96 shows almost the same crystal structure as POL150 and POL200, and they all have similar chemical compositions. The (400) projections of three pollucites with the interplanar spacing of 3.41 nm were also clearly observed in TEM analysis. The microstructure of POL96 seems like a snowball, and it has a much smoother surface than the POL150 and POL200. In Si-MAS NMR, the POL150 and POL200 samples have much more Si linkage than the POL96 due to more Si dissolution at higher temperatures (150 and 200°C vs. 96°C). This synthesis method is simple and cost-effective. Thus, we can develop structurally stable pollucite to treat much larger volumes of liquid wastes containing Cs-137.

Keywords: Cesium, Mordenite, Clinoptilolite, Pollucite, 96ºC, Immobilization

A Study on Mockup Test of Reactor Vessel Internal Segmentation

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After permanent shut down of Kori unit 1 (K1), the first commercial nuclear power plant in Korea, in last 2017, various decommissioning activities are expected. The segmentation of activated components, such as reactor vessel, internals, are classified as an important milestone of the whole project. The process attracts lots of attention in terms of radiation protection, waste management, and technical difficulties. The mockup test of the reactor vessel internal segmentation is essential for the process development. The mockup test enables the optimization of the process, verification of radiation protection and waste management plan.

The reactor vessel internal of K1 is Westinghouse 14 x 14 2loop type model. Many components are combine each other using fastener bolt. The thermal shield is cylindrical shape. Since reactor vessel internal of K1 is different from other types, the unique process and tools are suggested to achieve the safe and economical decommissioning.

Various segmentation tool and process are studied, such as band saw, disc saw, contact acr metal cutting (CAMC), etc. Since many parts of the internal is combined using fastener bold, they need to be unfastened to minimized the process duration. The coarse and fine segmentation process follows. The detailed process and comparison of the segmentation process is studied.

In this paper the reactor vessel internal segmentation process will be discussed and lesson learned from mockup test of the component will be also discussed.

Keywords: Kori unit1, Decommissioning, Reactor Vessel Internal, Mockup test

A Study on Radiation Protection During Reactor Vessel Segmentation

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During the decommissioning process, it is required to prepare the suitable safety plan in terms of radiation protection and industrial safety. The reactor vessel is one of the important component. It is a massive and relatively highly activated component. It had been exposed to large amount of neutron for a long operation period. The reactor vessel contains relatively large amount of radioactive nuclides, compared other large components.

Although the in-situ reactor vessel segmentation strongly relies on the remote cutting tool, there are various circumstances that workers need to be get close to the reactor and/or cutting tool. The representative regions are operating floor, upper cavity area, lower cavity area, nozzle area, and in-core room.

During the reactor vessel segmentation various works are expected in that regions. The representative expected work list in operating floor are assembly and optimization of cutting tool, installation of remote control tower, and work preparation. The representative expected work list in upper cavity are segmentation, withdrawal of segments, and inspection of cutting tools. The representative expected work list in lower cavity are installation of heating, ventilating, and air conditioning (HVAC) unit, segment packaging, package preparation.

In this paper, the radiation exposure to worker during reactor vessel segmentation process is evaluated. The evaluated radiation exposure dose will be used as a basic information for radiation protection and implement of ALARA principle.

Keywords: In-situ, Radiation Protection, ALARA, Decommissioning

A Study on Secondary Gaseous Waste From Reactor Vessel Segmentation

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The segmentation of activated components, reactor vessel and internal, is one of the primary process in the decommissioning of nuclear power plant (NPP). In terms of ALARA principle implementation, it is important to minimize the radioactive waste generation and spread.

The oxy-propane torch, representative thermal cutting method, enables rapid cutting process and minimizes process period. The process relies on the metal melting process. The heat energy from melted metal propagates to the neighboring metal. Since the process relies on the metal melting process, the oxy-propane torch approaches to the outer part of RV. The melted carbon steel, has lower melting temperature compared to stainless steel, provides thermal energy to the inner stainless steel cladding and melts them.

The thermal energy from melted metal and oxy-propane torch contributes the evaporation of materials. The gaseous materials are generated during cutting process. The heating, ventilating, and air conditioning unit (HVAC) maintains negative pressure in the cutting region and removes the gaseous materials from the cutting process.

In this paper the generation and characteristics of the gaseous materials are studied. The physicochemical properties, including diameter, elements, etc., will be discussed from the reactor vessel segmentation mockup test.

Keywords: ALARA, Decommissioning, Oxy-propane torch, Aerosol

A Study on Mockup Test of Reactor Vessel Segmentation

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The permanent shut down of Kori unit 1 opens new era in back-end nuclear cycle. The segmentation of activated components is one of the most important and difficult process in decommissioning. The neutron from the nuclear fuel activates various components. Since the reactor and internals are closely located to the fuel, they are considered as relatively highly activated materials. The segmentation of them need to be carefully controlled and monitored to satisfy the ALARA principle.

The segmentation process of reactor vessel is composed of preparation process, primary segmentation process and finishing process. During the preparation process, insulation separation, safety injection nozzle cutting, pre-cut, and inlet/outlet nozzle cutting process are implemented. The in-situ remoted segmentation of reactor vessel has advantages in process duration, accuracy of the cutting process, and control of heavy components.

The oxy-propane torch, representative thermal cutting method, enables the rapid cutting. Many process parameters, including applied power, cutting speed, nozzle to component distance, etc., are systematically controlled and studied. Additionally, the minimization of dross and effective separation of aerosol are studied to achieve the effective control of secondary waste from the segmentation process.

In this paper, the reactor vessel segmentation optimization will be discussed and lesson learned from mockup test of the component will be also discussed.

Keywords: Segmentation, In-situ, Reactor vessel, Mockup test

Neutron Activation Calculations for the Shielding Window of DFDF Hot Cell at KAERI

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About 26 kgU of spent nuclear fuels are stored in the DFDF (DUPIC Fuel Development Facility) hot cell of IMEF (Irradiated Materials Examination Facility) at KAERI. The components, equipment, casks, and structure of DFDF hot cell become activated when irradiated by the neutrons emitted from the spent nuclear fuels. Because the neutron flux density is expected to be very low, radioactivity concentrations by neutron activation may be within clearance criteria level. In DFDF hot cell, the five shielding windows are installed, and also can become activated by the neutrons from the spent nuclear fuels. In this study, we calculated the radioactivity concentrations by neutron activations for the shielding windows to verify whether the concentrations meet clearance criteria level, which is needed with respect to decontamination and decommissioning.

ORIGEN-ARP, MCNP, and FISPACT computer codes are used to calculate neutron source terms, neutron flux density, and radioactivity concentrations, respectively. The geometry of shielding windows in DFDF hot cell is modeled and the neutron source is assumed to be located homogeneously in front of the window with the volume of 50x50x50 cm^3. Because the activation level is heavily dependent on the depth, we calculate the neutron flux density by 5 cm. Using the neutron flux density for each section, the radioactivity concentrations are calculated using FISPACT. It is assumed that the spent nuclear fuels are stored for 30 years and radioactivity concentrations are calculated for cooling time of 30 days. The ratio of the radioactivity concentration to clearance criterial level is calculated in the end.

Ce-141 and H-3 are identified as major nuclides by activation of the glass, but it is confirmed that the radioactive concentration are extremely low because of low neutron emissions from the spent nuclear fuels. Fe-55, Mn-54, and Fe-59 are identified as major nuclides by activation of the cast iron frame, but it is also confirmed that the radioactive concentration are extremely. In this study, we confirm that the radioactivity concentration in the glasses and iron frame sufficiently meets the clearance criteria level and they can be considered as non-radioactive waste and reused. Even though the radioactivity level by neutron activation is very low because of small amount of neutrons from spent nuclear fuels, neutron activation may become more important problem for storage of large amount of spent nuclear fuels or neutron sources such as Cf-252 and Am-241/Be.

Keywords: Activation, Spent nuclear fuels, Shielding window, FISPACT, MCNP, Clearance

Self-propelled Magnetic Illite Microspheres for Active Removal of Radiocesium

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Radioactive liquid wastes containing ^{137}Cs is inevitably produced from the operation of nuclear facilities, and leaks or spills of the radioactive liquid wastes can cause contamination of environment. Although liquid waste treatment technologies have been developed to reduce the radiological impact of radiocesium on the environment, ion-exchange and membrane technologies are not suitable for selective removal of Cs from liquid wastes containing various competing ions, such as K^+, Na^+, Ca^{2+} and Mg^{2+}. Furthermore, such technologies require a large-scale water treatment facilities which should be decommissioned after the completion of waste treatment. As a result, such water treatment processes can produce a large amount of secondary wastes including facility decommissioning wastes and separation materials binding radiocesium and various competing ions. For these reasons, in-situ remediation technologies, which can selectively remove Cs from the liquid without special equipment, have received a great attention.

We report the illite-based Janus micromotors which can selectively and actively remove Cs from liquid waste by self-propelled motion. The illite microspheres containing iron oxide nanoparticles were prepared by a scalable spray-drying method, and a half surface of the microsphere was decorated with catalytic Pt layer. The highly Cs-selective frayed edge sites contained in the natural illite enabled selective removal of radiocesium from the simulated liquid wastes containing K^+, Na^+, Ca^{2+} and Mg^{2+}. The bubble propulsion of micromotors in the presence of hydrogen peroxide fuel increased the Cs adsorption kinetics, and the magnetic micromotors could be easily separated from liquids under external magnetic field.

Keywords: Clay, Illite, Cs, Adsorbent, Micromotor

Emulsion-based Synthesis of Hollow Silica Microspheres

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Since the discovery of mesoporous silica materials in 1992, numerous silica materials with uniform mesopores have been prepared via surfactant-templated method. Emulsion-based synthesis of porous particles also drawn much attention due to its relatively-benign synthesis conditions. In our present study, amine-based surfactant was used to stabilize the oil emulsion and preparation of the hollow silica sphere. Tetraethylorthosilicate (TEOS) was used as a silica source. Water and ethanol were used as a mixed solvent in the reaction mixture containing less than mmol% range of hydrochloric acid, and therefore the reaction rate was very slow. Surface area of the hollow mesoporous silica via acid-catalyzed surfactant-templated synthesis was > 900 m^2/g with pore volume of > 0.90 cm^3/g. Pore size was ~ 3.5 nm. Oil-in-water emulsion was formed in the reaction mixture with TEOS and a hydrophobic co-surfactant. Size of the oil-in-water emulsion determines the size of the hollow spheres. Emulsion droplet size depends on the relative amount of TEOS, the structure-directing agent, and co-surfactant. It also depends on the mixing methods such as ultrasonication. High-speed mixers can be used to decrease the droplet sizes down to nanometer ranges. It is promising to have such a controllable hollow inner space that various sizes of active ingredients or waste substances are accommodated inside of the hollow spheres through the small mesoporous channels with high surface areas.

Keywords: Emulsion, Mesoporous, Silica, Microsphere, Surfactant-templated synthesis

Monte Carlo Characterization of Depth Distribution of Neutron Activation in the Bio-shield of Concrete Depending on Distance From Reactor Vessel

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In 2017, Kori Nuclear Power Plant Unit 1 (Kori-1) was decided to be permanently shut-down and deemed to be dismantled. Radioactive concrete makes up a major portion of radioactive waste during dismantling nuclear power plant. Since the radioactive matter takes an extremely low portion in weight of the whole concrete shield, recycling of the radioactive concrete can be profitable. For example, the dismantled concrete wastes of TRIGA MARK III (Research Reactor 2) was 2,003 tons, among which 1,710 tons were non-radioactive and 3.3 tons were self-disposable. In this study, Monte Carlo simulation has been performed to characterize the radioactivity varying with the depth in the concrete bio-shield.

Simulation was carried out using the MCNP 6.2 code. Reactor types have various distance between bio-shield and reactor vessel: 132 cm for Kori-1 and others for diverse PWRs (5 cm, 17 cm etc.) Radioactivity distribution in the Kori-1 bio-shield was simulated by simplifying the nuclear power plant geometry of core, barrel, bypass, thermal shield, down-corner, and vessel. The components and density of concrete bio-shield were assumed to be the same. The neutron source flux was referred to the Kori-1 data available from an earlier study. Calculation was made by varying distance from the reactor core to the inner surface of concrete bio-shield from 30 cm to 200 cm. With FMESH tally and FM tally, concentrations of radioactive nuclides in concrete as bio-shield was calculated by MCNP 6.2 code. After that, Bateman equation was used to get nuclides' equilibrium concentration with MS-EXCEL. From the result, we could estimate the volume and activity of bio-shield activated by neutron from reactor core, depending on air space from vessel to bio-shield.

In Kori-1 bio-shield, total activity of 5.346×10^3 [Bq/g] was located at 304.75 cm from the center of reactor core and attributed to the radionuclides including ^{60}Co (99.6792%), ^{152}Eu (0.3112%), ^{154}Eu (0.0106%), and ^{134}Cs (0.0001%). Calibration curves for cutting depth using the clearance thickness (specific activity of less than 0.1 Bq/g) of bio-shield concrete are presented for width of air space.

Furthermore, it is useful to proceed the study of dismantling other nuclear power plants in Republic of Korea in the future and calculate the dose of radiation worker during dismantling the nuclear plants for radiation protection.

Keywords: Radioactive Concrete, MCNP, Clearance Boundary, Nuclear Power Plant

Development of Human Error Simulation Stress Model in Nuclear Power Plant Decommissioning Activity

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In this study, the influencing factors of stress were selected through literature review and definition of stress, one of the PSFs of RPVI decommissioning work. And the weighting factor of each selected influencing factor was derived. Based on this, the human error simulation model stress part was developed. The stress experienced by workers associated with RPVI decommissioning works can be defined by "time stress" of having to carry out the work defined within the planned time period and "cognitive and workload stress" of the amount of work to be carried out, intensity, and complexity of work. Most of the HRA literature explains that the time pressure according to time to diagnose and act on an event causes stress, and the magnitude of the stress is influenced by work skill and knowledge. And in the literature on task stress of nuclear power plant operators, 13 factors were selected as factors for stress by synthesizing the contents of the stress-related literature. Based on these, eight factors were selected, categorizing the stress factors on the RPVI decommissioning work into work stress related to the work, team stress due to the work done by each team, and personal stress due to individual workers. And the weighting of the eight influencing factors of selected stress was derived. The weighting of influencing factors was derived using the Fuzzy-AHP method suitable for quantification of expert qualitative decision-making. As a result of the derivation, the weighting of task characteristics and fatigue among the eight influencing factors was found to be high as 0.2205 and 0.176. This means that the characteristics of the task performed by the worker and the fatigue of the worker have the greatest impact on the stress of the worker performing the RPVI decommissioning work. Lastly, the eight factors that influence the stress of the worker of the RPVI decommissioning work and the weighting for each influencing factor were applied to the human error simulation stress models. The human error simulation stress model can easily grasp the relationship and importance of stress factors, and if appropriate management of the derived influencing factors can reduce stress on workers. However, in actual work, it may be difficult to manage the entire eight influencing factors, so it is expected that priority management of the most important factors will result in efficient management of worker stress.

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Keywords: Stress, Influencing Factor, Weighting, Human Error Simulation Model

A Study on the Mass Balance Verification of the Decontamination Reagent Regeneration Process by the Cation Ion Exchange in the Magnetite Oxide Reductive Decontamination by Oxalic Acid

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As part of the nuclear power plant primary system chemical decontamination, we studied the reductive decontamination characteristics of magnetite oxide, which accounts for most of the corrosion oxides subject to decontamination in the nuclear power plants. Oxalic acid was used as a reductive decontamination reagent, and the chemical composition of residual decontamination reagent, decontamination products and pH depending on the amount of magnetite decontamination were studied experimentally. The result of experiments were compared with theoretical calculations in the following two cases; with regeneration of decontamination reagent by cation ion exchange and without regeneration.

The experimental results of the residual oxalic acid concentration and pH according to the magnetite decontamination were consistent with the following theoretical prediction within the experimental error range.

(i) Residual oxalate concentration without oxalate regeneration

$$Cres.wt\_ox = Co\_ox - 1.33\ C\_FeC_2O_4$$

(ii) Residual oxalate concentration with oxalate regeneration

$$Cres.w/o\_ox = Co\_ox - 0.33\ C\_FeC_2O_4$$

(iii) pH change depending on residual oxalic acid concentration

$$C\_H^+ \quad = 0.4475*Cres\_ox^{\wedge}0.8611$$

Keywords: NSSS chemical decontamination, Magnetite reductive decontamination, Oxalic acid, Regeneration of decontamination products

DeCoEx : Decommissioning Waste and Cost Analysis Tool Using Excel

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In Korea, the first commercial nuclear power plant (NPP), Kori 1, is under preparation for decommissioning and the rest of NPPs are expected to be decommissioned once their design lives expire. For each NPP decommissioning cost are required to be reviewed and updated in every two years, so many decommissioning cost evaluation programs using bottom-up approach are being developed with more user-friendly UIs. However, these programs still have limits in data analysis and require users to have thorough understandings of the decommissioning cost evaluation algorithm. In this paper, a newly developed user friendly NPP decommissioning waste and cost estimation program, DeCoEx, is introduced and its advantages are explained.

DeCoEx is developed using mostly commonly used calculating tool, Microsoft Excel, and incorporated algorithmized cost evaluation procedures of NPP decommissioning. DeCoEx consists of 15 database sheets, 1 plant inventory input sheet, 1 contingency and additional cost input sheet, and 4 result sheets. Despite there are numerous sheets, all of them are displayed and can managed in a single sheet, called module sheet, so users require less time moving around the sheets to compose the database. Furthermore, as the sheets are divided in detail, user can easily analyze and manage any aspects that could affect decommissioning cost, such as decommissioning strategy, plant inventory, unit cost of work activity, labor cost, etc.

In comparison with DeCAT, which is the most frequently and commercially used program for NPP decommissioning cost estimation in Korea, DeCoEx displays all the required data in a single screen and allows user to access all the database and every steps of cost calculation, so users can readily understand the process of decommissioning cost estimation and easily compose the database. Furthermore, DeCoEx uses pivot table function of Microsoft Excel which allows users to customize and display desired results in single screen, so compared to DeCAT, which require extra procedure to arrange the data for analysis, users require less effort and directly analyze desired field, such as decommissioning cost, waste disposal volume, waste type, etc.

In this paper, the newly developed NPP decommissioning cost estimation program, DeCoEx, and its main functions are explained. It is expected that the program can be used independently for cost evaluation, as well as for verification of other existing decommissioning cost evaluation program.

Keywords: Nuclear Power Plant Decommissioning Cost Evaluation, DeCAT, DeCoEx, Program Guide

6분과

방사선환경 및 안전 (Oral)

Quantum Dot-based Monolithic Plastic Scintillator for Gamma Detection

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Recently, radiological characterizations and measurements in nuclear decommissioning are of potential and important issue. In-situ Object Counting System (ISOCS) is a representative in-situ measurement system that can accurately evaluate the level of contamination of decommissioning wastes and sites. Although this system has high accuracy, it has disadvantage such as cost, difficulty in management, size, and weight. Therefore, it is necessary to improve the measurement methodology of inorganic and plastic detector or to develop the new detection sensor in order to be used for in-situ measurement. In this regard, the development of new scintillators has been a potent and necessary subject of research of years. In this study, it will present a quantum dot-based plastic scintillator to detect Cs-137 contaminated soil. Thus, a monolithic typed plastic scintillator loaded with a quantum dot (0.2 wt%) was fabricated and measurement experiments were performed in 10 Bq/g of ^{137}Cs contaminated soil. As a result of the experiment, the relative efficiency was improved by about 20-30% compared to the commercial scintillator (EJ-200). Based on the results of this study, the possibility of plastic detector loaded with a quantum dot was confirmed. In Future, the photopeak is observed in the energy spectrum using the plastic detector, it will be possible to develop a large plastic detection system with a resolution similar to that of an inorganic scintillator.

Keywords: Plastic scintillator, Quantum dot, Gamma detection, Efficiency, In-situ measurement system

Development of Beta-spectrometer Comprising Multi-wire Chamber and Plastic Scintillator

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It is essential to measure the amount of positron generation in the positron transport technique for positron annihilation spectroscopy. In the positron moderation process by hitting a tungsten mesh or foil to generate a slow positron beam, since only one of 10^3 to 10^4 positrons are released, the measurement of the initial positron generation is necessary. However, in most positron beam-emitting environments such as research reactors and linear accelerators, gamma rays are also emitted. Thus, it is necessary to selectively measure beta rays using a beta-spectrometer. In addition, this study aims to contribute to neutrino physics by experimentally measuring an antineutrino energy spectrum of ^{238}U. The antineutrino energy can be determined by the measurement of beta-ray energy. For measurements on a ^{238}U antineutrino energy spectrum to be useful, the detector must have a high gamma-ray interference removal rate.

The beta-spectrometer of this study consists of the gas-filled multi-wire chamber for counting the beta-ray and the plastic scintillation detector for energy spectroscopy. The beta-spectrometer detects beta-ray using a coincidence method, which selects signals that only occur simultaneously within a specific time using timing logic for two signals.

In this study, to evaluate the performance of developed spectrometer, the energy spectra were measured with the radionuclide sources (^{137}Cs and ^{207}Bi) emitting internal conversion electrons and gamma rays. In addition, a ^{60}Co source was used to evaluate the gamma ray removal ratio of the beta-spectrometer, and the measured value with coincidence mode and the measured value without coincidence were compared. As a result, the gamma-ray removal ratio of the spectrometer was 99.12%. To reconstruct the beta spectrum from the measured spectrum, Monte Carlo simulations using the Geant4 toolkit were performed to calculate coefficient related to the effect of gamma rays. Based on the obtained spectrum, it was confirmed that the energy resolution and intrinsic internal conversion electron detection efficiency of the beta-spectrometer were 10.2% at 1 MeV and 44.2%, respectively.

Keywords: Beta-ray, Gamma-ray, Spectrometer, Detector, Coincidence

Preliminary Study on Gamma Ray Analysis via Artificial Neural Network Method for a Seawater Monitoring System

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Busan metropolitan city has installed and been operating four 3 in. x 3 in. NaI (Tl) gamma detectors near the Busan metropolitan city coast to monitor radioactivity in seawater. This study was conducted to compare the detection performance of conventional method and machine learning analysis method, we proposed an algorithm based on an artificial neural network (ANN) and calculated MDA of 137Cs in seawater for measurement situations. To make data for training, validating and test the ANN, we have made two data sets. The one data set is background data that is achieved from real measurement data i.e. natural background from seawater monitoring system, the other data is simulated data that performed Monte Carlo simulations with the MCNP code using a seawater monitoring system. We made virtual 137-Cs gamma spectrum with various activity and combined the real background spectrum. We used 25,000 virtual data to train ANN 4,000 virtual data to validate the trained ANN, and 1,000 virtual data to test the trained ANN. To compare the performance of ANN we checked the error of results with various 137-Cs activity that range from 1 Bq/L to 50 Bq/L. The result shows if the true 137-Cs activity is about 10 Bq/L with 15 min measurement time the ANN predicts true value with high accuracy (below 10%).

Acknowledgements

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Keywords: Artificial neural network, Seawater monitoring system, Gamma-spectrum analysis, NaI(Tl) scintillation detector

Uncertainty Analysis of the Clearance Levels of SRS-44 Model Using Monte Carlo Simulation

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Nuclear facilities generate various levels of radioactive wastes. Clearance is to release the radioactive wastes that have radiologically trivial risks from regulatory control. The International Atomic Energy Agency (IAEA) presented the radioactivity mass concentration levels in IAEA Safety Guide No. RS-G-1.7 (RS-G-1.7) in 2004. And the methodology for the derivation of clearance levels was published in the IAEA Safety Reports Series No. 44 (SRS-44) in 2005. The clearance levels have been widely used in many countries and have applied to the IAEA Safety Requirement Part 3 (GSR Part 3), which is the current International Basic Safety Standard (BSS).

However, the parameters applied in the model have fluctuation by the country or site-specific features and it makes uncertainty. If it is specified the significant parameters and calculate the uncertainties of the model, we can derive the major parameters and can also derive the radionuclides that have relatively small margin before the clearance. Especially the increasing numbers of nuclear power plants that are in the state of permanent shutdown or decommissioning worldwide, the radioactive wastes for clearance within various conditions are expected to increase. Therefore, it is necessary to review the assumptions in the SRS-44 model and quantify the uncertainties to find out the model's marginal extent.

Primary radionuclides for clearance wastes were selected from former studies and calculated clearance levels from every pathway in the SRS-44 models to define their dominant pathway. The major parameters were derived by performing a deterministic sensitivity analysis to the parameters used in those pathways. The distributions of the major parameters were assumed by the literature study and for each of the parameters was performed Monte Carlo random sampling simulation. The sampling has performed for 10,000 times using Cristal Ball simulation tool.

In this study, the major parameters affect to the dominant pathway were derived and calculated the marginal extent of doses for the radionuclides for 95% reliability level. This study can be used to support the preliminary clearance process for sort out the parameters or radionuclides that need to be concerned.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS), granted financial from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No.2106025).

Keywords: Radioactive Waste, Clearance, Clearance level, Sensitivity Analysis, SRS-44

A Preliminary Study on the Factors to Reduce Radiation Exposure to Workers During Decommissioning Overseas Nuclear Power Plants

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Even if nuclear power plants (NPPs) is permanently shut down, some radiation sources can give radiation exposure to workers continuously, proper exposure management for workers should be carried out when decommissioning NPPs. In Korea, there is no experience in decommissioning except for research reactors, so it is necessary to investigate cases of overseas decommissioning - NPPs to find out how to manage the exposure dose of workers.

In United States, the exposure doses of decommissioning workers are systematically managed through accumulated experience. In this study, a literature survey was conducted to determine that it could be used to prevent radiation exposure of workers in advance and to establish appropriate protective measures through analysis of factors affecting radiation exposure to decommissioning workers. Several main factors were derived so that they can be used when establishing.

Factors that affect the exposure dose in overseas decommissioning NPPs can be divided into four major categories. The first factor is safe storage. If delayed decommissioning (safe storage) is implemented, it can cause radioactive decay of nuclides with a short half-life and finally it can significantly lower the level of radioactivity. The second one is full system chemical decontamination, which can remarkably reduce the radiation level in the primary system. The third one is underwater and remote cutting. This has the effect of blocking the occurrence of internal exposure of workers due to radioactive dust that can occur during the cutting work. The last one is the number of plant loops or output. However it did not have a significant effect on the dose reduction.

In conclusion, it was found that four factors affecting workers' exposure during decommissioning were safe storage, full system chemical decontamination, underwater and remote cutting operations, and the number of loops and outputs. Since there is no experience in evaluating the radiation exposure to workers during decommissioning in Korea, this study can be useful for the radiation protection of workers when establishing a NPP decommissioning plan in the future.

Keyword: NPP decommissioning, Radiation protection, Radiation workers, Exposure dose

Radioactive Effluents Released From Korean Nuclear Power Plants and the Resulting Dose to the Public

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In operating nuclear power plants (NPPs), the generation of radioactive effluents is inevitable, and radioactive effluents must be thoroughly managed to protect humans and the environment. Radioactive effluents can be classified into gaseous and liquid radioactive effluents. In this paper, the exposure pathways from gas and liquid radioactive effluents, the amounts of radioactive effluents, and the exposure dose to the public are analyzed. In addition, it was confirmed that the exposure dose to the public living around NPPs was lower than the annual dose limit of 1mSv per year in accordance with the Enforcement Decree of the Nuclear Safety Law.

Gases and liquids radioactive effluents are discharged into the environment depending on their characteristics. As a result of analyzing the exposure pathways of gaseous and liquid radioactive effluents, the main routes of gaseous and liquid radioactive effluents were the release to air and the sea, respectively. Analysis of average effluents discharged from Korean NPPs over the past three years (the Year 2017-2019) reveals that tritium accounts for the primary nuclide of gas and liquid radioactive effluents. Gaseous tritium was released the most at 109TBq from Wolseong NPPs, and liquid tritium was discharged the most at 70.8TBq from Saeul NPPs. Analyzing the exposure dose for the last three years as an average value, the exposure dose (99.49%) from gaseous radioactive effluents accounts for a higher proportion than the dose (0.51%) from liquid radioactive effluents. In addition, the main nuclide in the exposure dose to the public was carbon-14. Carbon-14 in gaseous radioactive effluents was identified as the main contributor with 2.80×10^{-2}mSv/y at Wolseong NPPs.

In conclusion, it was found that carbon-14 in the gaseous radioactive effluents is the main contributor to the dose to the public living around NPPs; however, the resulting dose is much lower than the annual dose limit of 1mSv.

Keywords: Radioactive effluents, Exposure pathway, Exposure dose, Tritium, Carbon-14

Long-term Evaluation of Desorbed ^{137}Cs Transport Through Porous and Fractured Aquifers Below the Artificial Lake

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The 2D cross-sectional aquifer model was developed to simulate the behavior of desorbed ^{137}Cs by the seepage of groundwater beneath the artificial lakes storing the potable water resources. For a systematic investigation of ^{137}Cs transport, various scenarios were considered with reported ranges of distribution coefficient (K_d), local temperature conditions (basal heat flow and lake-bottom temperature) affecting on ^{137}Cs desorption rate, heterogeneous (coupling K and K_d) and fractured aquifer properties. The characteristics of ^{137}Cs plume represented by average plume concentration and the migration rate of the mass center were assessed, and the health risk of chronic exposure for humans was calculated using discharged ^{137}Cs mass at the downstream of the artificial lake.

The ^{137}Cs transport in base-case (K=8.64×10^{-2} m/day and K_d=10 ml/g) revealed that approximately 0.06% of the initially released ^{137}Cs arrived downstream; the maximum migration rate was 4.4×10^{-4} m/day and 4.2 mSv/y of the annual radioactive dose appeared at 220 years. However, the notable effect of varying K_d on the migration rate (2.6×10^{-5} ~ 8.1×10^{-4} m/day) and annual dose (1.0×10^{-10} ~ 120 mSv/y) indicates that the K_d is the most critical parameter to control the transport of ^{137}Cs plume. The effect of temperature conditions on the migration (3.7×10^{-4} ~ 5.5×10^{-4} m/day) was relatively insignificant, but the decreased desorption rate at cold temperature further reduced the annual dose (0.4 ~ 9.8 mSv/y). In addition, 50 realizations of heterogeneous K and K_d were generated to evaluate the influence of the aquifer heterogeneity. The migration rates (1.4×10^{-5} ~ 2.0×10^{-4}) and annual dose (9.0×10^{-4} ~ 2.8 mSv/y) of ^{137}Cs plume showed more irregular and wide variation in the coupled K and K_d heterogeneities. Finally, continuum-based fracture networks generated from the discrete fracture network were applied to evaluate the effect of the fractured aquifer and revealed the importance of fracture orientation and connectivity.

Keywords: ^{137}Cs contamination, Risk assessment, Aquifer heterogeneity, Fractured aquifer

Development of a Compact Boron Removal Device Which Minimizes the Spent Resin Generation Based on Desalination Technology

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Currently, regulation trend of boron has been stricter in the world. NPP's in Korea also are taking it very seriously to release boron into the ocean because of the environmental pollution of boron. This boron removal technology is being developed to satisfy the domestic boron emission regulation due to global interest in terms of the environmental protection. Since boric acid is a molecular substance and exists in molecular form below pH 7, there is a limitation in removing it with the existing ion exchange resin and RO membrane in the non-ionized state. In general, EDI technology or pH control technology using chemical agents are considered, but there are disadvantages such as environmental pollution due to the use of chemicals and economic costs for using the separators. EDIXR (Electrochemical Deionization Exchange Resin) technology, which is currently being developed, is based on ELIX (Electrochemical Ion Exchange) technology developed for the purpose of removal and recovery of boron at the Ringhal nuclear power plant in Sweden. From its technological aspects, the technology is believed to satisfy various sizes, operation method and flow rates required by domestic related regulations and nuclear power plants sites. In this study, the project focused on chemical model of the boric acid removal performance and mechanism with actual boric acid solution, based on lab scale EDIXR device. For the measurement of boron concentration, a potentiometric titrator was used in this study.

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Keywords: Boric acid, Boron, Electrochemical, Electrochemical deionization exchange resin

6분과

방사선환경 및 안전 (Poster)

In-situ Radioactivity Measurement for a Radioactive Wastewater Decay Tank

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Using unsealed radioactive isotopes for industries or hospitals produces radioactive wastewater. In order to comply with the Nuclear Safety Act, radioactive wastewater should be stored in a dedicated tank and should be drained into the public environment only after confirming that it makes negligible radioactive pollution. Since decay depends only on time, the storage period of radioactive wastewater is determined by its half-life, and decayed wastewater sufficiently to satisfy the standards for drainage is discharged after measurement of radioactivity by gamma spectroscopy with a semiconductor detector or gamma counter. However, these tools are expensive equipment, and a large cost burden for radioisotope users. So, in this study, a method of continuously measuring radioactivity by installing a detector on a storage tank was employed to save cost.

To do this, in this study, a regional monitoring system was employed that continuously measures the radiation level in a fixed place. In this system, 1.5 inch NaI(Tl) scintillation detector was used to measure radioactivity in a storage tank, and activity conversion factor (ACF) for ^{18}F was calculated. NaI(Tl) scintillation detector is commonly used in studies measuring radioactivity in water. When measuring underwater radioactivity, water acts as a shield and affects the measurement radius; thus, the change in ACF was observed while increasing the size of the storage tank containing radioactive wastewater. The change in the counted value for various sizes of the tank containing radioactive wastewater was calculated using MCNP 6.2. As a result of the calculation, when the volume of the tank was over 32 L, the ACF was fixed at 2.3075×10^4 Bq•s•m^{-3}•counts^{-1}.

Because the system used in this study must prove that the radioactivity of wastewater in storage tank is less than the drainage standard, the minimum detectable activity (MDA) of this measurement system must be less than that of radioactive wastewater. To prove this, the MDA of the measurement system was calculated to be 6.254×10^6 Bq•m^{-3} for one-hour measurement, which was smaller than 1×10^7 Bq•m^{-3} prescribed by Nuclear Safety Act that the public is predicted to reach the dose limit due to draining of ^{18}F containing wastewater to the natural environment.

Keywords: NaI(Tl), Wastewater, Decay tank, Activity conversion factor, Nuclear Safety Act

Measurements of Gamma-ray Source Position and Dose Rate Using Plastic Scintillation Optical Fiber Bundle Detector

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After the Fukushima Daiichi Nuclear Power Station accident (2011), a large amount of radionuclides were released into the environment. In a large-scale nuclear accident, it is essential to investigate and characterize the radionuclide in question, such as the activity and geographic/topological distribution of the radionuclide, in order to understand the stabilization and decontamination of the radionuclide. Monitoring of dispersed radionuclides during and after decontamination is also necessary to evaluate the decontamination process. Surveys of contaminated areas can be divided into aerial and ground surveillance of radionuclides. Although aerial surveys can quickly detect a wide range of radionuclides, they are generally less accurate than ground surveys. Ground surveys include the use of backpack detectors, vehicle-based detection systems, and more. Most commercial ground surveying systems use inorganic scintillators such as NaI:Tl, or common surveyors such as Geiger-Muller counters. However, these detectors are not suitable for large area surveys because they can only provide point data and it is difficult to obtain the correct geometry in large volumes or contaminated areas. In this case, the contaminated area can be mapped by a kriging method that interpolates the data from the measurement points with the data at positions not sampled by weighted linear estimation using adjacent data. In this study, we manufactured a 5 m long position-sensitive plastic scintillation optical fiber (PSOF) bundle detector consisting of a sensing probe, two photomultiplier tubes, two high-speed amplifiers, and a digitizer. The seven PSOFs in a bundle were used as sensing probes to estimate the position of the gamma-ray source, and uncollimated ^{137}Cs sources were used as gamma-ray emitters. One-dimensional source positions were measured, and their spatial resolutions were estimated by full width at half maximums. Moreover, the dose rates according to the distances were measured and compared with the results obtained using a Fluke model 421B Ion Chamber Survey Meter.

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Keywords: Gamma-ray source position, Plastic scintillation optical fiber, Spatial resolution, Full width at half maximum, Optical fiber bundle detector

Fast Temperature Compensation for NaI(Tl) Spectrum Using Count-Center Method

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The NaI(Tl) radiation monitor is a radiation measuring instrument for discriminating radionuclide, and is widely used in various radiation fields. NaI (Tl) generates luminescence proportional to the incident radiation energy, and radionuclides can be identified through analysis of intrinsic radiation energy determined by the measured amount of light. However, the weakness of the NaI (Tl) system is that the amount of light changes due to environmental factors such as temperature, which makes it difficult to determine the nuclide. A commonly used temperature compensation technique is an automatic calibration method using natural nuclides (K-40, Tl-208), and in order to use this technology, the total energy peak of the natural nuclides must be measured to an extent that can be detected. This requires a few minutes or more of measurements. In this study, a temperature compensation algorithm that is faster than the automatic calibration method using natural nuclides was introduced and its performance was verified. The CountCenter method proposed by us calculates the center channel, which is the center point of the spectral distribution. It is a monitoring technology, a concept similar to a mass-center. CountCenter can be calculated as a distribution of counts versus channels. The spectrum according to temperature shows a similar pattern to the gain change, and temperature compensation is possible by monitoring the CountCenter change according to temperature and adjusting the gain according to the fluctuation value. Using a temperature/humidity chamber for a 3 inch x 3 inch cylindrical NaI (Tl) detector, the spectral change and temperature compensation method according to the temperature change from -20 degrees to 50 degrees were evaluated. The K-40 full energy peak of the background signal was monitored, and K-40 showed fluctuations up to 40 channels, which means an energy fluctuation of about 120 keV. In the natural nuclide automatic calibration method, K-40 (1460keV) and Tl-208 (2614keV) peaks were detected with a 90% probability when measured for 5 minutes, and temperature compensation was performed by adjusting the gain based on the channels of the two peaks. On the other hand, when the CountCenter method was used, the change was detected within 20 seconds, and temperature compensation was performed by applying the CountCenter changed ratio. After that, the temperature compensation was determined using the artificial nuclide/natural nuclide peak. In both methods, temperature compensation was performed normally, and when the CountCenter method was used, calibration was possible in a short time.

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Keywords: NaI(Tl), Temperature compensation, Decommissioning site, CountCenter

A CdS/ZnS Quantum Dot-based Plastic Scintillator

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The period of decommissioning due to the permanent shutdown of commercial nuclear power plants is approaching in Korea. During decommissioning, an assessment of the contamination level of decommissioning wastes must be performed. Concrete wastes account for more than about 80% of the decommissioning wastes, and the nuclide of interest in concrete wastes is ^{60}Co. In this study, a plastic scintillator including CdS/ZnS quantum dots was fabricated. And, performance evaluation was performed using a ^{60}Co point source. After adding quantum dots of high atomic number and fluorescent materials such as PPO (2,5-diphenyloxazole) and POPOP (1,4 di[2-(5phenyloxazolyl)] benzene) to the plastic matrix, plastic scintillators were manufactured through thermal polymerization. In addition, cutting and polishing processes were performed for the transparency of the plastic scintillator. The manufactured plastic scintillator was connected to PMT and MCA (Multi-Channel Analyzer) to analyze the measurement performance. In this experiment, a ^{60}Co point source of 236.95 kBq was used, and data were collected after setting the distance between the source and the detector as 20, 50, and 100 mm. In addition, the plastic scintillator was analyzed based on the Compton energy region because the photopeak was not observed and showed a continuous spectrum according to the detection characteristics. As a result, the relative efficiency of the plastic scintillator including CdS/ZnS was higher than compared to the commercial scintillator And the relative efficiency was increased to by 5-30% as the measurement distance was increased. This is because the dead time increases as the distance between the radiation source and the detector is increased. The dead time is a factor that has a close relationship with the detection efficiency, and it is the time required to process the signal of the incident particles. During the dead time, the information of the incident particles is lost. Therefore, it means that the incidence rate to the detector must be sufficiently low to reduce the dead time. Based on this study, the possibility of using a quantum dots loaded plastic detector as a gamma measurement system was confirmed, and it is expected to be used in medical fields as well as decommissioning field.

Keywords: Plastic scintillator, CdS/ZnS, Gamma detection, Decommissioning, In-situ measurement system

Measurements and Estimation of Beam Qualities of X-ray Spectra With CdTe Detector

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To realize and validate continuous reference filtered X radiation according to ISO 4037-1:2019, two following methods are possible. One is the measurement of half value layer (HVL) and the other is the spectrometry. In 2020, KHNP-CRI has established 14 X-ray fields by means of HVL measurements. However, it was necessary to estimate beam qualities such as mean energy and spectral resolution (FWHM) by spectrometry to determine the effects of the spectrum due to changes in the tube voltage or thickness of addition filtration. In this work, X-ray spectrum measurements and analysis were performed on three X-ray fields (HK30, HK100, WS300).

In order to produce HK30, YXLON MG165 was used and MG325 was used for the HK100 and WS300. Spectral measurements were performed with X-123 CdTe detector (AMPTEK Inc.,). The energy calibration of the spectrometer was conducted using ^{133}Ba to cover the energy range up to 300 keV. X-ray intensity was controlled by lowering tube current and installing the tungsten collimator with an aperture 0.5 mm in diameter in front of the additional filtrations to minimize dead time losses and pile-up effects. And if an additional collimation was needed, EXVC X-ray collimator kit (AMPTEK Inc.,) was installed in front of the spectrometer. The detector was positioned in the center of the radiation field size at a distance of 1 m from the target. X-ray spectrum and background was acquired during (500 – 700) second at 700 H.V. For converting the pulse height spectrum into the spectral fluence, the response matrix of spectrometer was calculated by use of MCNP6.2 code and the obtained spectra were unfolded with HEPROW code.

The pulse height spectrum of HK30 and HK100 were obtained at 1.0 m distance by installing additional EXVC X-ray collimator with an aperture 1 mm. The tube current was 0.07 mA and 0.05 mA which leads to a dead time of 9.81%, 3.72% respectively. In the case of WS300, it was found that X-ray was penetrated EXVC X-ray collimator and measured. To minimize X-ray intensity without EXVC X-ray collimator, the measurement was performed at 2.5 m distance and tube current was lowered to 0.03 mA which leads to a dead time of 7.68%. As a results of analysis on beam qualities of unfolded spectrum, the mean energy of HK30 and HK100 and WS300 were 19.7 keV, 56.2 keV and 189.5 keV respectively. The spectral resolution of WS300 was 67.2%. The beam qualities of HK30 and HK100 were in good agreement with ISO standards but out of WS300. This seems to have affected the spectrum by adjustment of tube voltage and additional filtration. A further study on dose conversion coefficient would be conducted.

Keywords: X-ray, Spectrometry, CdTe detector, Unfolding, HEPROW

A Linear Range for the Assessment of Dose Rate Using Airborne Gamma-Ray Spectrometer Based on CZT Detectors

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According to lesson learned for Fukushima Daiichi nuclear power plant accident, diverse survey platforms should be applied to areas with variety dose rate levels for efficient consequence management, such as characterizing radiations and making deposition map of radioactive materials. Especially, gamma-ray spectrometers in the field with a high dose rate must have enough capability to accurately measure incident radiations and safely store the results in the survey system. The MARK-A1 (monitoring ambient radiation of KAERI – airborne platform #1) based on CZT (CdZnTe) detectors was developed to show a high performance in the case of high dose rate level. Diverse commercial CZT detectors with different sensor volumes can be easily mounted to MARK-A1 through a BNC connector. In this study, two CZT sensors with a volume of 10x10x5.0 (CZT500S) and 15x15x7.5 (CZT1500) mm^3 were used to determine their linear range in assessing ambient dose rate in the areas with a high dose rate level. First, the irradiation experiments using a ^{137}Cs source were conducted to estimate the performance of two gamma-ray spectrometers. The energy spectrum was then measured during 120 s in 21 irradiation conditions from 0.001 to 20 $mGy{\cdot}h^{-1}$. All control units of MARK-A1, such as MCA (multichannel analyzer), Bluetooth interface, and PC, keep the basic operation with the airborne gamma-ray spectrometer. From the spectrum analysis depending on 21 irradiation conditions, the ambient dose rate was calculated by multiplying total count rate and dose conversion factor, G(E), depending on used CZT sensors. Gamma-ray peaks from ^{137}Cs were also analyzed to check the variation of energy resolution according to the dose rate level, by representing FWHM and FWTM (full width at half maximum and tenth maximum) at 662 keV. A linear range of two CZT detectors was finally determined by comparing irradiated and measured dose rates as well as considering the relatively constant range of FWHMs and FWTMs at 662 keV.

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Keywords: linear range, dose rate, gamma-ray spectrometer, CZT detector, Irradiation

The New Modular Gamma Probes for Finding out Hidden Nuclear Materials in a Restricted Area

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The history of nuclear activities and the accountancy of special nuclear materials should be reported and also verified within the process of denuclearization. However, it realistically assumes that some hidden facilities or materials still exist. KINAC has developed a small-sized searching device associated with unmanned aerial vehicles (UAV) for finding out the hidden facilities and materials. The device is called Probe and expected to provide more accurate and intense radiological information by special operation method. Probes are carried by a UAV and spread out over the suspicious area. The radioactive hotspot is then pointed out by the difference of the strength of signals proportionally changed by the distance from the source.

Last year, the 1st prototype Probe has shown the potential for radioactive surveillance and isotope identification but also concluded that design optimization is required to improve cost efficiency. Thus, we newly designed the 2nd Probe using plastic scintillators.

The new 2nd Probe was designed to achieve detection efficiency and less weight of the system. For these, the internal structure of the system was fully changed to multi-stacked PCBs. Since the plastic scintillator was used, it is impossible to adapt the multi-channel analyzer but the weight of the system was much less than before. The reduced weight in the system was preserved by increasing the scintillator size. And the larger-sized scintillator brings better detection efficiency.

The telecommunication method was also changed. the LoRa, which was adapted in 1st Probe, is the most efficient method for using a battery system, however, the stability and multi-connectivity were not enough to use in our system. Thus, the new telecommunication system based on the Wi-Fi and Mesh architecture was adapted in the 2nd Probe.

There were various developments, and we would like to present all the developments of the Probe system at the conference.

Keywords: Probe, Plastic scintillator, Detection efficiency, Small Gamma-ray detectors

Optimization of the Analytical Process for Radiostrontium in Soil

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Strontium-90, which exists at a low level in the environment, is a beta-emission source of continuous spectrum and is mixed with other radionuclides, making its quantitative analysis difficult. Therefore, it is necessary to increase the specific radioactivity of ^{90}Sr through an enrichment process, and a chemical separation process to isolate it from other radionuclides that affect the analysis is required.

An analysis procedure for ^{90}Sr measurement in environmental soil samples was conducted using Sr resin, and the results were measured using a low-level liquid scintillation counter, Quantulus 1220. The recovery was estimated using a HPGe semiconductor detector. For the traceability of measurement, 0.05-0.3 g of ^{90}Sr certified standard material (KRISS) was added to a 20 mL Teflon vial, and the volume was made to 10 mL using distilled water.

In the Cerenkov mode, the counting efficiency was determined to be 76.69%, the minimum detectable activity (MDA) of ^{90}Sr was 0.7 Bq L^{-1}, and the recovery was 47.13%. It was confirmed that the ^{90}Sr measured in the collected soil was less than the MDA.

Keywords: ^{90}Sr, Cerenkov, Liquid scintillation counter, Sr resin, Recovery

Development of a Radiation Monitor Using Indoor Location Tracking Technology for Nuclear Power Plant Dismantling Sites

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Only the United States, Germany, Japan, and Switzerland have experience/know-how of nuclear power dismantling in the world, and three countries excluding the United States have only one to three nuclear power dismantling experiences, so there is insufficient data on nuclear power dismantling technology. The first commercial nuclear power plant decommissioning procedure is underway for Kori 1. For the development of nuclear power plant decommissioning technology, the dismantling work of Kori 1 will be very important data, and a lot of information generated during the nuclear power plant decommissioning process must be obtained. The dismantling process of nuclear power plants proceeds in the order of shutdown-preparation for dismantling-decontamination-waste treatment-site restoration, and the analysis of radiological information for each process is very important data. The existing RMS system is highly likely to be unavailable because power/communication is used through the disassembly process, and it is difficult to evaluate the installation of a large area as it is installed/operated mainly for core facilities. In this study, we designed/developed a radiation monitor using indoor location tracking technology at the nuclear power plant dismantling site. The developed radiation monitor enables indoor location tracking technology by applying UWB technology (Ultra Wide Band). It is possible to relocate and move in real time, and to collect real-time radiological information through wireless. UWB measures TDoA through communication with more than 4 ANCHORs that know the location and calculates the distance between ANCHORs. The calculated distance uses the triangulation method to calculate the coordinates of the equipment with low MMSE. The radiation monitor to which UWB technology was applied was equipped with a detector of the GM tube to measure the dose rate in real time. After attaching 5 ANCHORs to a laboratory with a width of 13m and a length of 8.3m, the location tracking technology of the radiation monitor was evaluated. As a result of evaluating the sensing position and the actual position for an arbitrary point, it was confirmed that indoor position tracking is possible with an error of up to 0.5 m. Based on this, as a result of moving and measuring the radiation monitor, it was easy to collect information on the dose rate by moving distance and to determine the location of the source through the contour map. If this technology is used, it is expected that radiographic information collection and safety evaluation standards for nuclear power plant decommissioning sites will be established.

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Keywords: UWB, Nuclear power plant, Decommissioning site, Location tracking

Development of High-pressure Ion Chamber System for Nuclear Power Plant Decommissioning Site

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This study develops and evaluates the HPIC used at the nuclear power plant decommissioning site. The high-pressure ion chamber can monitor a wide range of radiation dose rate suitable for use in the nuclear power plant decommissioning site, and because it uses a current measurement method, it is easy to measure in a high radiation field. The pressurized ion chamber was manufactured in a circular shape with a standard of 10 pies, and a high-purity argon gas (99.9999%) was injected into the interior and filled with a pressure of 25.5 bar to perform the test. The radiation source used for the test was Cs-137 having an energy of 662 keV, and an isotropic test was performed to evaluate the linearity of the energy according to the radiation intensity and to determine whether the range of 360 degrees can be measured. Energy linearity evaluation was performed using an energy of 3 μSv/h-700 mSv/h, and the isotropic test was performed at intervals of 30 degrees from 0 to 360 degrees. In order to measure the intensity of radiation in a wide range of energy, an ultra-precision current meter was developed, and since the current meter is vulnerable to noise in the pico-ampere (10-15A) range, the current range is divided into three stages (low level, medium level, and high level). A circuit was developed to measure. As a result of the test, it was confirmed that saturation did not occur even at a certain counting rate or higher because the high-pressure ion chamber does not have dead time. These characteristics can measure a wide range of dose rate from low-level radioactivity intensity to high-level radioactivity intensity, and are considered to have excellent detection performance in a nuclear power plant decommissioning environment. In addition, as a result of measuring the current in the range of 12 points in total in the isotropic test, it was confirmed that the current was within 95% in all ranges and had excellent isotropy.

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Keywords: HPIC, Nuclear power plant, Decommissioning site, Current circuit

Development of a Radiation Direction Detecting System for Searching Orphan Sources

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This study establishes a radiation source direction detecting system to find orphan radiation sources in the environment. The detectors used NaI:Tl based scintillation detectors, consisting of a total of eight. The eight detectors are composed of four at the top and four at the bottom, with the top four facing counterclockwise, the bottom four facing clockwise, and facing all four directions (±X, ±Y axis). The result value is derived by using the distance calculation formula and the detected value for each detector, and the direction to the orphan source is calculated through the ratio of the distance (circular equation) and the detected value. The calculation method used the Gauss-Jordan elimination method using Python. The test was carried out through the Monte-Carlo based MCNP simulation. The radiation source was placed 40cm away from the center of the system, and simulations were performed in the 360-degree direction at 45-degree intervals. Based on the simulation results, an actual detection system was fabricated and evaluated, and the test was performed in the same method as MCNP using Cs-137 (@662keV). As a result of the calculation, it was confirmed that the error rate for the direction in both conditions was within 1 to 8%, but it was difficult to estimate the distance, and an additional study will be conducted to evaluate the distance.

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Keywords: Orphan source, NaI:Tl scintillator, Radiation direction detecting, Python

A Study on Probabilistic Methodology for Constructing Inverse Tracking of Source Term Estimation

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In the event of an accident, radioactive materials leaked into the environment are diffused and deposited in a wide area depending on weather conditions, and may cause exposure to the general public in the short and medium to long-term after the accident. And then it is more effective to protect residents by predicting accident information quickly and accurately. The environmental radiation monitoring facility in the area around the Nuclear Power Plants (NPPs) provides only some information, such as the ambient dose rate of the area located. Therefore, it is difficult to use it for predicting the source terms of radioactive material in the event. Recently, the development of air diffusion evaluation technology applying Artificial Intelligence (AI) technology has been reported abroad, and the diffusion evaluation of harmful substances is being performed. Bayesian inference is one of the machine learning techniques that makes accurate inferences by combining a conditional variable model obtained results using a probability distribution and a numerical characteristic of experience or prior knowledge about the conditional variable. In the field of meteorology for many years, many studies have been used to estimate the location of the fundamental emission source that discharges harmful substances in the atmosphere. However, research at NPPs considering the terrain features of Korea is insufficient. Bayesian inference methodology includes Markov chain Monte Carlo (MCMC), Sequential Monte Carlo (SMC), Differential Evolution Monte Carlo (DEMC), and polynomial chaos expansion (PCE) techniques. In this study, we described the composition of the inverse tracking algorithm for source term estimation using the MCMC.

Keywords: Probabilistic Methodology, Inverse Tracking, Source Term, Inference, MCMC

Statistical Analysis of Radioactive Effluent Data From Korean Nuclear Power Plants Using R Program

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When nuclear power plants (NPPs) are operated, radioactive materials are discharged to the environment, and each country made to comply with the ALARA principle as a concept of optimization in order to reasonably reduce the effects of radioactive materials released to the environment while operating NPPs. The International Atomic Energy Agency (IAEA) establishes radioactive effluent discharge limits for facilities and activities primarily to optimize the protection of the public. The discharge limits are set higher than the actual discharges and must be set within specific dose limits. The IAEA and OECD/NEA mentioned that margins of operational flexibility and headroom should be considered in order to establish reasonable discharge limits. This requires an assessment of the previous experience and realistic/actual discharge of similar facilities, and should be determined according to the characteristics of the facilities or activities that affects the discharge.

Therefore, in order to reasonably set the discharge limit, the actual discharge data of 26 NPPs in Korea were analyzed by NPPs and nuclides group. In consideration of the characteristics of each NPP and each nuclide group, a statistical analysis was performed using the Mann-Kendall Trend Test and the R Program. As major results, it can be seen that the liquid and gaseous radioactive discharges decreased due to the improvement of nuclear fuel integrity and treatment system performance, and the efforts of the operator to reduce environmental discharge. Among the effects of operational history (event), the discharges of gaseous fission products increased due to nuclear fuel defects. In addition, the characteristics of radioactive effluents released from NPPs according to the installation of liquid and gaseous radioactive waste treatment systems were lower at the PHWR equipped with a tritium removal facility compared to other PWR. In addition, liquid fission and activation products decreased sharply in NPPs equipped with waste evaporators.

Through this study, it is possible to statistically analyze the radioactive effluent data using the R program by the major operational history (event) of each NPP, and the change in characteristics of discharges according to the installation of waste treatment system.

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Keywords: Radioactive Effluent, Discharge Limit, Operational Flexibility, R Program, Operational History, Radioactive Waste Treatment System

Evaluation of Gaseous Effluent Concentration Limits Based on the ICRP Publication 103, 134, 137 for ^{3}H, ^{60}Co, ^{131}I, ^{137}Cs, and $^{234, 235, 238}U$

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In general, the effluent concentration limit (ECL) of radionuclides from nuclear facilities can be derived by taking into account the dose limit, effective dose coefficient, respiration rate, and activity time of the public. In fact, NSSC (Nuclear Safety and Security Commission) of Korea notified ECLs based on ICRP Publication 60, considering the DAC (Derived Air Concentration) concerned with inhalation of 1 μm AMAD of radiation workers and the difference of dose limit, the effective dose coefficient, respiration rate, activity time between radiation workers and the public.

Subsequently, the International Commission on Radiation Protection (ICRP) published ICRP Publication 103 (The 2007 Recommendations of the ICRP) and ICRP Publication 130, 134, 137 (Occupational Intake of Radionuclides). As a result, the effective dose coefficient, respiration rate, absorption type, etc. have been modified.

In this study, gaseous ECLs based on ICRP Publications 103, 134, 137 were evaluated for major nuclides such as ^{3}H, ^{60}Co, ^{131}I, ^{137}Cs, and $^{234, 235, 238}U$. The range of ratios of ECL based on ICRP Publication 103, 134, and 137 to the current ECL based on ICRP Publication 60 and 119 is 0.98 ~ 3.33 for ^{3}H, 0.54 ~ 0.95 for ^{60}Co, 1.15 ~ 1.36 for ^{131}I, 0.87 for ^{137}Cs, and 0.40 ~ 2.06 for $^{234, 235, 238}U$. In addition, ECLs for new compounds or absorption types of ^{3}H, ^{60}Co, ^{131}I, ^{137}Cs, and $^{234, 235, 238}U$ have also been newly evaluated. From a conservative point of view, the current ECL based on ICRP Publication 60 should be changed to the new ECL based on ICRP Publication 103, 134 and 137.

Keywords: ECL (Effluent Concentration Limit), ICRP 60/103/134/137, $^{3}H/^{60}Co/^{131}I/^{137}Cs/^{234,235,238}U$

Review of Overseas Dose Constraints Establishment Status and Rationale

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When a nuclear power plant is operated, radioactive effluents which can lead to exposure to residents are generated. In accordance with international organizations, exposure to residents should be managed by establishing dose constraints for the radiological effects of radioactive effluents. However, research to introduce the concept of dose constraints is insufficient in Korea. In this study, the definition of dose constraints of international organizations, the status and rationale of overseas dose constraints were investigated.

International organizations have suggested definition and establishing methodology of dose constraints. ICRP defined the dose constraints as the upper limit of the dose expected to be received by the planned operation of a specific source (e.g. nuclear power plant, etc.). IAEA suggested to set the dose constraint as the top of the distribution by analyzing the dose distribution during operation. In addition, IAEA suggested to set the dose constraints as the average dose of the critical group.

In United States, 250 μSv/yr was suggested in terms of dose based on the acceptable cancer risk for nuclear facilities. In UK, 150 μSv/yr was suggested for waste disposal facilities based on the risk acceptable to the public. In Canada, 50 μSv/yr was suggested in terms of higher level than the general public exposure distribution measured for 10 years for nuclear facilities.

In this study, the definition of dose constraints of international organizations and the status and rationale of overseas dose constraints were investigated. International organizations have defined the dose constraints as the upper limit of the dose expected to be received by the planned operation of a specific source, and recommended that it be set as the upper end of the dose distribution. In overseas countries, dose constraints were set based on simple conversion of the degree of risk or based on a level of exposure distribution. In Korea, when setting the dose constraints, exposure distribution should be derived by reflecting the domestic radiation usage, and the dose constraints should be set based on this distribution to meet international organization recommendation.

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Keywords: ICRP, Dose constraint, Radioactive effluent, Regulatory system

Analysis of Hazard Scenario of Nuclear Facility for Preparation of Radiological Environmental Impact Assessment Report (RER)

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To obtain the license of nuclear facility from commission, it is necessary to prepare radiological environmental impact assessment report (RER). The RER is composed of radiological assessment of operation and from hazard on the facility. For radiological assessment of operation, the environmental impacts of the operation of nuclear facility should be evaluated in a quantitative way as possible. For radiological assessment from hazard, potential environmental impacts from hazard should be evaluated. For this, the first thing to do is to analyze the hazards scenarios that may occur to the facility. Therefore, this study analyzed hazard scenarios that may occur to spent nuclear fuel processing facilities. Referred of documents of IAEA and USNRC, hazard scenario that could arise for a facility were analyzed by divided into external events and internal events. External events include natural external events, such as extreme weather conditions, and human induced events, such as malevolent vehicle assault. In accordance with 10 CFR Part 20 and 10 CRF Part 72, safety classification of spent nuclear fuel processing facilities is RW-IIc (Non-safety). Therefore, external events of these facilities are made up earthquake and wind, precipitation (rain, snow). Considering the climatic characteristics of Daejeon, wind and precipitation (rain, snow) exclude from external events. Internal events should be selected in consideration of the radiation risk of the facilities, example of internal events is pipe breaks, load to drop, internal explosions and fire. Internal events of spent nuclear fuel processing facilities are made up load drop, fire of filters, explosion of components and so on. In this study, the hazard scenario used in the preparation of RER of the spent nuclear fuel processing facilities were analyzed. This results will contribute to radiological assessment from hazard. Furthermore, it will be used to prepare RER of nuclear facility, including the spent nuclear fuel processing facilities.

Keywords: Radiological environmental impact assessment report (RER), Radiological assessment of hazard, Hazard scenario, Safety classification

The Trends of Radiation Level Inside Reactor Hall and Exhaust Air on HANARO

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This paper shows the trends of radiation level in the HANARO reactor hall and the exhaust air from the reactor hall. Those trends measured every hour from 2010 to 2019. HANARO is an Open-pool type research reactor. So, radioactive materials, which can affect the level of radiation, from the reactor pool can be released into the reactor hall. During the HANARO operation, radioactive materials such as Xe, I, and Kr may be released. In this paper, we will consider the radiation level of the reactor hall and exhaust air through the ventilation system which be composed of the MODE and HEPA filters in the 2010s.

During this period, there were reactor shutdowns (long-term shutdown due to Seismic Retrofitting Construction), White Emergency Alarm (it is incident of impact on reactor hall) and Reactor Pool Surface High-Radiation Alarms. The long-term shutdown period was 2014 to 2017 about two and a half years. It was enough time to decay. Because half-life of I-131 is 8.02days, Xe-133 is 5.24days and Kr-88 is 2.84days. It is possible to predict the variation of radiation level in the reactor hall after the reactor is stopped. White Emergency Alarm occurred in February 2011, because activated floater that be installed in the reactor pool floated on the pool surface. Another trip which caused by High-Radiation Alarm occurred in April 2012. Activated air bubbles inside the capsule leaked out and came up to the surface of the pool. The other is a manual trip by the operators due to malfunction of Hot Water Layer System in December 2017. It increased the reactor pool surface radiation level. By conducting the radiation trend in abnormal state, we can prepare for the situation what will be happen.

According to analysis of radiation level of inside reactor hall and exhaust air in abnormal state, it will be used to predict next conditions. Also it shows that all seems to be very much under control.

Keywords: HANARO, Reactor hall, Exhaust air from reactor hall, Trends of Radiation level

Comparison of Conservative and Realistic Dose Evaluation Based on the Outflow Rate of the Spent Resin Treatment Facility

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Spent resin generated from heavy water reactors requires treatment as the radioactivity concentration of ^{14}C exceeds the criteria for ILW (intermediate level waste) disposal. Therefore, the treatment facility that does not generate secondary waste and desorbs ^{14}C using a microwave suitable in terms of treatment efficiency is under development. The maximum amount of spent resin that can remain inside the facility is 600 kg, and the actual remaining amount is 125 kg. Therefore, in this study, conservative (600 kg) and realistic dose (125 kg) evaluation of the spent resin treatment facility with a treatment capacity of 1 ton/day was performed for comparison and analysis. The external dose was evaluated using the VISIPLAN code, and the internal dose was evaluated in consideration of APF (assigned protection factor) and resuspension rate. Dose evaluation was performed in units of 10 % in the range of the outflow rate of 10% to 100%. In addition, it was assumed that the worker removed the spilled spent resin for 1 hour.

The dose range of worker without APF was derived as 2.55E-01 mSv/y ~ 1.92E+01 mSv/y in conservative evaluation and 5.32E-02 mSv/y ~ 4.00E+00 mSv/y in realistic evaluation. In the case of considering APF, the dose range was derived as 2.01E-01 mSv/y ~ 7.27E-01 mSv/y in conservative evaluation and 4.19E-02 mSv/y ~ 1.51E-01 mSv/y in realistic evaluation. Despite the conservative evaluation that did not consider APF, it was confirmed that the dose was less than 20 mSv/y, which is the average annual dose limit of worker regardless of the outflow rate. However, in terms of ALARA, it is expected that worker should wear an air-purifying respirator to reduce the worker's dose.

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Keywords: ^{14}C, Dose assessment, Spent resin, Radiological safety

Comparative Analysis of Methodologies for Dose Assessment due to the Radioactive Effluents: USNRC Regulatory Guide 1.109 and UNSCEAR 2016 Methodology

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The best objective of the control radioactive effluents discharge from the NPPs during the normal operation of Nuclear Power Plants (NPPs) is to minimize the radiological impact to the public and environment under the As Low As Reasonably Achievable (ALARA) principle in the most of countries. To assess the impact of radioactive effluent, the several methodologies to calculate the radiation dose are widely used such as US Nuclear Regulatory Commission (USNRC) Regulatory guide 1.109 which has been used in Korea and U.S.

In 2016, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) published "UNSCEAR 2016 Report on Sources, Effects, and risks of ionizing radiation" and the Annex A of this report suggested the methodology for estimating public exposures due to radioactive discharges to the atmosphere and the ocean during the normal operation of NPPs. In the case of radionuclides' dispersion in the ocean, this methodology applies the compartment model (divided into the local, regional and global) which differs from the current assessment model in Korea. Since the different assumptions of each assessment tool may cause the various results, it is necessary to observe the effects of those assumptions.

Therefore, this study compares the differences between the USNRC Regulatory guide 1.109 model and the UNSCEAR 2016 Report Annex A methodology. In addition, this study simulates the dispersion of the radionuclides by modeling the compartments with AMBER code based on the UNSCEAR 2016 methodology. This study introduces the application of the compartment models in the dose assessment of discharged radioactive effluents in Korea.

Acknowledgements

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Keywords: Radioactive effluents, Discharge, Dose assessment, Compartment model, UNSCEAR 2016 Report, AMBER code

Application of ALARA to Radiation Workers in Reuse of Containment Building of Kori Unit 1

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The external radiation specific derived concentration guideline levels for Kori Unit 1's containment building were obtained using the RESRAD-BUILD since the external radiation is generally considered as the main exposure in the evaluation of nuclear facilities. Using VISIPLAN 3D ALARA Planning Tool, the safety of workers was confirmed as the total dose received will be lower than the effective dose limit allowed for radiation workers. ^{60}Co was identified as the radionuclide that produces the highest amount of external radiation and receives the most shielding.

Acknowledgements

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Keywords: Decommissioning, Containment building, RESRAD-BUILD, VISIPLAN

Dose Assessment of the Radioactive Waste Disposal Facility by Applying Representative Person Concept

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Nuclear Safety and Security Commission (NSSC) sets a notice to manage radiation dose to the public resulting from the radioactive effluent from radioactive waste disposal facility. The International Commission on Radiological Protection (ICRP) published the ICRP 103 report and recommended applying the concept of representative person to assess radiation doses to the public. However, such concept was not fully accepted in the current regulation system. The maximum exposure individual concept instead of representative person has been used for radiation protection. The new concept is expected to be reflected in regulation system in the future. The objective of this study is to assess radiation dose of radioactive waste disposal facility applying representative person concept.

In order to assess radiation dose applying representative person concept, exposure scenarios and exposure pathways should be defined. Therefore, six exposure scenarios were set up. In the case of exposure pathway, six exposure pathways were set up based on domestic regulatory guidelines. ICRP recommended setting up a location where actual people reside when selecting critical group. Therefore, in this study, a cadastral map was used to select the location where actual people live near radioactive waste disposal facility. Also ICRP recommended that 95 percentile values should be used for the dominant habit data and lower values should be applied for other habit data. Therefore, in this study, 95 percentile values were used for the dominant habit data. Also average values were used for other habit data.

In this study, the concept of representative person was applied to assess the radiation dose assessment of radioactive waste disposal facility. To assess representative person dose exposure scenarios, exposure pathways, critical group candidates, and habit data were established. There are eight critical group candidates were set up based on cadastral map. In the case of habit data ingest crops and leafy vegetables were dominant habit data. The result of radiation dose assessment for the representative person dose showed 5.62×10^{-5} mSv/y. The results are approximately 40% lower than 9.93×10^{-3} mSv/y which is assessed as maximum exposure individual concept. The result of this study will be used as a prior study for the introduction of representative person concept recommended by ICRP 103 in Korea in the future.

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Keywords: ICRP 103 recommendation, Dose constraint, Representative person, Radioactive waste disposal facility

Analysis on the Domestic and Foreign Regulatory Standards for Radiation Exposure in NORM Industries

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Naturally occurring radioactive materials (NORM) contain natural radionuclides and cause radiation exposure. Each country has been establishing and managing regulatory standards for NORM industries suitable for industrial conditions. The objective of this study is to investigate and analyze the domestic and foreign regulatory standards for radiation exposure in NORM industries to review radiation protection trends. We investigated regulatory standards of IAEA, Canada, France, and South Korea.

In IAEA GSR Part 3, the clearance levels were set as 10 Bq/g for K-40 and 1 Bq/g for U and Th series. IAEA recommended that graded approach be taken to the regulation control, so that the application of regulatory requirements is commensurate with the radiation risks. Canada sets the dose criteria for NORM industry workplaces with classification and carries out graded management. Investigation threshold and NORM management threshold were set 0.3 mSv/y to the public and workers. Where doses to workers or public may exceed this value, a site-specific assessment should be carried out. If assessed dose to the public and workers exceeds 0.3 mSv/y, it is classified as NORM management target. Also, dose management threshold and radiation protection management threshold were set 1 and 5 mSv/y to the workers, respectively. In France, if the amount of radioactive material is less than 1 ton, it is managed the same as the clearance level in GSR Part 3. If the amount of material is more than 1 ton, dose assessment to workers should be conducted. If the dose to workers exceeds 1 mSv/y excluding radon, it is required notification and environmental impact assessment. In Korea, NORM exposure is managed by the act on protective action guidelines against radiation in the natural environment. The domestic standards are 10 Bq/g for K-40 and 1 Bq/g for other natural radioactive nuclides included in raw materials and by-product. If the radioactivity concentration exceeds the standard, it is required registration and managed collectively.

As a result of analysis of domestic and foreign regulatory standards for NORM, in other countries, regulatory management for NORM industries is carried out based on graded approach. However, in Korea, if the radioactivity concentration is higher than the criteria, it is classified to notification and managed collectively. Considering the low exposure of the NORM industries and effectiveness of regulation control, it is necessary to manage exposure to NORM industries with a graded approach in Korea. The results of this study can be used for managing radiation exposure at NORM industries.

Acknowledgements

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Keywords: NORM, Radiation exposure, Graded approach, Radiation protection, IAEA

Review of the Domestic and Foreign Safety Management of Radon in Indoor Air

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Radon can be absorbed into the human body and cause internal exposure. Therefore, radon management in the air is necessary for public safety management. The objective of this study is to review the domestic and foreign status of radon safety management in indoor air.

In the United States, the U.S. Environmental Protection Agency (EPA) manages radon in indoor air through the National Radon Proficiency Program (NRPP). EPA suggested 148 Bq/m^3 as the recommended standard for indoor radon concentration. During 1988-2013, more than 23 million radon measurements were performed, and radon reductions were performed on more than 1.24 million houses. In the UK, Public Health England (PHE) maintains a radon action level of 200 Bq/m^3 and a target level of 100 Bq/m^3. If the results of radon measurement show a value between the action level and the target level, the reduction measurements should be considered. If the radon measurement is below the target level, the reduction measurements do not need to be considered. In Ireland, the Radiological Protection Institute of Ireland (RPII) performs radon management based on the “Radiological Protection Act”. In this law, the recommended standard for indoor radon concentration is 200 Bq/m^3 for houses and 400 Bq/m^3 for workplaces. In Korea, radon management is carried out by the Nuclear Safety and Security Commission (NSSC) and the Ministry of Environment (MOE). NSSC manages substances containing natural radionuclides including radon, based on the “Act on Protective Action Guidelines against Radiation in the Natural Environment”. MOE manages radon in the air based on the “Indoor Air Quality Control Act”. In this law, the recommended standard for indoor radon concentration is 148 Bq/m^3 for multi-use facilities such as libraries and art museums. In addition, the recommendation standard of radon for new apartment houses was suggested as 200 Bq/m^3.

In this study, we investigated the domestic and foreign status of safety management of radon in indoor air. In each country, government agency was designated and managed as a radon management agency. The domestic recommendation standards of radon were found to be at a similar level when compared with the foreign standards. The results of this study can be used as basic data for the safety management of radon-induced radiation.

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Keywords: Radon, Safety management, Reference level, Radioactive concentration

Water Quality and Resin Performance Improvement by Removing CO_2 From Air Injection of Generator Stator Cooling System

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Commercial large generators generate some heat during electricity production and the system heat must been eliminated effectively to sustain system materials. The generator’s stator cooling method injects forcibly cooling water into the stator to absorb heat inside the stator, and discharges it to the outside environment. At this time, the quality standards of the stator coolant are set to minimize corrosion of system materials according to the copper material of the cooling system. The water corrosion characteristics of copper materials are minimum in the range of dissolved oxygen (DO) from 1 ppm to 2 ppm, pH is from 7 to 8, and electrical conductivity of water is 0.5 μ/cm or less, and the stator cooling water is provided. To this end, a certain amount of air dissolved in water of the cooling water storage tank for the supply of dissolved oxygen. Mixed ion exchange resin used to reduce the dissolved ion and electrical conductivity of cooling water.

However, there is approximately 400 ppm of carbon dioxide in the air, and when it is dissolved in water, it ionizes and indicates weak acidity. Ionized carbon dioxide also has a significant negative effect on the service life of the ion exchange resin.

In this paper, we evaluated the solubility of carbon dioxide in the stator cooling water and the load of ion exchange resin due to carbonate ions during the air saturation process. We evaluated the load of carbonic ion in case of compressed oxygen and compressed nitrogen injection method.

Alternatively, we proposed in designing carbon dioxide capture devices in the air when injecting natural air.

Keywords: Stator cooling system, Copper corrosion, Ion exchange resin, CO_2, pH

A Study on the Performance Evaluation of Boron Waste Liquid Separation/Concentration Using Electrochemical Techniques

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Boric acid (H_3BO_3) is used to control the reactor power in PWR's. Most of diluted boron contained water generated after the dilution operation is being recovered through the boron recovery system, but if the diluted water quality conditions are not satisfied to reuse, it is transferred to the liquid waste treatment system (LRS) to releasing to the ocean.

Korean standard nuclear power plant (OPR1000) applies an ion exchange resin and a reverse osmosis (RO) device to the liquid waste treatment without a waste evaporator. That means it is inefficient to remove the boron because it exists as a boric acid form under the low pH condition. Since the concentration regulation of boron in wastewater is becoming strict around the world, there is a need for a technology capable of additionally removing boron before discharging the boron contained wastewater.

Therefore, in this paper, in order to respond with the technical requirement of separating and concentrating boron in the waste monitor tank before final discharge, it is proposed a method to increase the boron removal efficiency by utilizing electrochemical deionization technology. In order to develop a technology to increase the boron separation and concentration efficiency, FEDI (Fractional Electrodeionization) and ELIX (Electrochemical Ion Exchange) technologies are developing. Using the advantages of the two devices, an experiment was conducted to determine the boron separation and concentration efficiency performance.

To compare the boron separation and concentration performance of the FEDI and ELIX devices, the concentration of boron in the waste monitor tank in the LRS was investigated. In addition, artificial contaminated water was prepared in consideration of conditions in which nuclides could enter the waste monitor tank due to the inflow of some contaminated water in the LRS. In order to compare the performance evaluation of the two devices, various experimental conditions were applied, such as comparison of the inflow concentration and flow rate of artificial contaminated water and the amount of current (V, A) using a DC power supply.

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Keywords: Boron, Boric acid, Electrochemical, Electrochemical deionization exchange resin

Generation Status and Management Prospect for Contaminated Water in Fukushima Nuclear Power Plant

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In 2011, Fukushima Daiichi Nuclear Power Plant units 1 to 3 had core melt down, and water to cool the reactor was continuously being injected. The cooling water, which has filtered out most of the Cesium and Strontium in the cooling water, passes through the desalination facility (Reverse Osmosis), so that the cooling water with relatively high contamination is sent back to the melted down reactors, and the cooling water with low contamination passes through ALPS, a multi nuclide removal facility. ALPS consists of a pretreatment facility and an adsorption tower. The adsorption tower consists of 14 replacement type adsorption towers and two column type towers. ALPS decontaminates 62 nuclides, but cannot filter out tritium (H-3). As of January 21, 2021, 1,243,131 m^3 of contaminated water was generated. To date, the storage capacity of contaminated water inside of Fukushima Daiichi Nuclear Power Plant is 1.37 million tons. The average specific radioactivity of tritium (H-3) in contaminated water is about 730,000 Bq/L, and the total radioactivity is estimated to be about 860 TBq. For reference, the total amount of tritium (H-3) generated in the reactor is estimated to be about 2,069 TBq. In the meantime, ocean discharge, evaporation, deep stratum injection, and long-term storage outside Fukushima Nuclear Power Plant have been considered as management measures for tritium (H-3). Fukushima Nuclear Power Plant decided to dilute contaminated water in storage on April 13, 2021 to a level of about 1,500 Bq/L, which is lower than the Japanese ocean discharge limit (60,000 Bq/L), and discharge it, and it will be implemented from 2023, two years later. Currently, there are Ru-106, Sr-90, I-129, C-14, Tc-99, etc. for nuclides that exceed the discharge limits in contaminated water, and they are planned to be purified again before discharge to the ocean.

Keywords: ALPS, Tritium, Contaminated water, Ocean discharge

Unplanned Release of Radioactive Materials to On-site Groundwater of Nuclear Power Plants in the USA

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The US Nuclear Regulatory Commission (NRC) defined an unplanned release as “The unintended discharge of a volume of liquid or airborne radioactivity to the environment”. The groundwater contamination due to the leak or spill of pipes and valves of the nuclear power plants (NPPs) containing liquid radioactive materials also included in unplanned release. According to the result of the NRC's investigation, the groundwater contamination by an unplanned release were occurred in 37 of the 65 operating NPPs until 2013. In most NPPs where unplanned release occurred, tritium was basically detected in on-site groundwater, and in some NPPs other radionuclides such as cesium-137, cobalt-60, nickel-63 and strontium-90 were additionally detected. Recently, the NRC is publishing a report on the status of tritium concentration for operating NPPs that have a history of unplanned release as described above. In this paper, we introduce the unplanned releases cases of radioactive materials to on-site groundwater of the three pressurized water reactors in the USA.

The Braidwood station reported three times of leak from circulating water vacuum breaker valves and one time steam release from turbine building as unplanned release events. Due to the events the maximum concentration of tritium in on-site monitoring wells and drinking water are 1,702 Bq/L and 59 Bq/L, respectively. The Callaway plant reported a leak of radioactive materials from air-relief valves during planned releases through the discharge pipeline as an unplanned release event. Due to the leak, the tritium samples ranged from 740 ~ 7,400 Bq/L. And also, radioactive cobalt and cesium were detected in the surface soil where the valves are located. The Indian Point found a hairline crack with moisture along the wall of the spent fuel pool. The crack was occurred due to the excavating the ground near the Indian Point Unit 2 fuel storage building. The properties of the crack’s moisture were same with spent fuel pool water. The tritium was detected in the crack’s moisture samples, and also during the continued investigation cesium-137, cobalt-60, nickel-63 and strontium-90 were detected in the groundwater. As a response of this event, the accessible areas of the spent fuel pool were inspected and six suspect areas coated with epoxy.

Although there are on-site groundwater contamination due to unplanned release in above three NPPs, it was confirmed that the level of radioactivity did not affect the residents.

Keywords: Unplanned release, Groundwater, Nuclear power plants, Radioactive materials

Establishment of Practical Minimum Detectable Activity Criteria of Tritium in Urine Sample for Nuclear Power Plant Workers in Korea

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Currently, the Korean Nuclear Power Plants (NPPs) adopt the Minimum Detectable Activity (MDA) criteria for tritium in urine samples as recommended by the American National Standards Institute (ANSI) N13.30 (0.37 Bq/cc). The ANSI committee considered selecting the criteria whether or not the values presented in the published literatures, such as International Commission for Radiation Protection (ICRP) publications, were reasonable in terms of health physics and achievable for laboratory. Therefore, it is necessary to consider the same conditions to set the MDA criteria for tritium at the Korean NPPs what the ANSI committee did.

The probability of internal exposure due to the tritium is relatively higher in pressurized heavy water reactor (PHWR) than that in pressurized water reactor since heavy water is used as both moderator and coolant in PHWR. Thus, the tritium monitoring for workers' urine sample, using a liquid scintillator counter, mainly conducted at PHWRs in the Korea. The period of the urine sample monitoring program is usually two weeks. Due to a large number of urine samples, approximately 20 000 urine samples in a year, the measurement time is set as 30 seconds or 1 minute per sample. Although the measurement time is short, the annual performance test results show that all performance index is satisfied with the acceptable range. However, the MDA of tritium (0.41 - 0.88 Bq/cc) is difficult to meet the criteria of the ANSI N13.30. A health physicist (HP) in the Korean NPPs controls the internal exposure dose of workers following the reference level, according to the dose level, such as screening level (0.02 mSv/y), recording level (0.1 mSv/y), investigation level (1 mSv/y) and medical intervention level (16 mSv/y). If a worker's exposure dose was expected below the screening level, the HP does not perform dose assessment. If the exposure dose is expected between the screening level and the recording level, the HP records the dose. Therefore, the NPPs urine analysis system should be able to measure the concentration of tritium in urine samples that are at least corresponding to the screening level. Furthermore, the internal dose assessment method for the tritium intake recommended by the ANSI N13.14 is applied to the Korean NPPs. With that method, the internal exposure dose corresponds to the screening level if 1 Bq/cc of tritium was detected in the urine sample of the worker under normal monitoring conditions (14 days).

Finally, it is estimated that 1 Bq/cc, which is a reasonable amount from the perspective of health physics and achievable in a current analysis system conditions of laboratory in the Korean NPPs, can be applied as a practical MDA criteria for tritium in the urine of the Korean NPP workers.

Keywords: Internal exposure dose, Minimum detectable activity, Urine analysis, Tritium

Development of Safety Shelter for Excavation Place Against Nuclear Accident

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In case of nuclear accident, the residents near to nuclear power plant shall be excavated for safe place for escaping radiation exposure.

Generally according to IAEA INES (International Nuclear Event Scale) class standard residents are excavated above INES class 4.

Even after escaping the high radiation dose areas residents shall be kept in safe residential condition against radiation exposure in excavation place.

In this regard ORION E&C develops the new idea design of safe shelter for securing the safety of resident place or building which accommodates the residents at excavation area.

The shelter type is moving house and local control type. The size is 8,000(L) x 3,000(W) x 5,000(H)cm and the weight is 1,4 tons. The shelter can be folded by three step panels according to the environmental conditions, and the driving unit is geared motor type which has rail and wheel as well as H beam and shape steel.

The main shelter materials consist of SS400 and form and can be protected against the contaminated air intrusion by adopting differential pressure system. Normally in excavation place the radiation exposure can be produced by air contamination condition rather than external radiation exposure. To protect air contamination and internal exposure by contaminated air inhalation the normal pressure in shelter is kept at 3,400mm H_2O (0.34 atm).

The shelter also has the fast air circulation system for circulating the air in shelter to reduce the contamination possibility.

This shelter can give the useful safety system in excavation place during nuclear accident and can be utilized at decommissioning site for covering container vessel to protect radiation release from decommissioning work site.

The covering of container vessel concept is vice versa concept for covering of excavation place in radiation protection point view because it protects the radiation release from in to out.

Keywords: Shelter, Excavation, Internal Exposure, Contaminated Air, INES

Equipment Improvement Case for Efficient Operation of Chemical Cleaning Wastewater Equipment

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As the number of years of operation of a nuclear power plant increase, sludge adheres to the tube sheet, tube support plate, and tube surface in the form of metal oxides on the secondary side of the steam generator. The deposition of corrosion products causes degradation of the tube and loss of heat transfer, which is a cause of impeding the stable operation of the steam generator. Such corrosion products are removed through lancing and chemical cleaning. Chemical cleaning to remove metal compounds deposited on the secondary system tube of the steam generator is removed by chelate bond using EDTA. Wastewater containing EDTA is treated using a pyrolysis method using high temperature because it is difficult with the existing treatment method.

The main components of the equipment for a pyrolysis are composed of an electric furnace, an oxidation furnace, a catalyst device, a heat exchanger, and a filtration device. Among them, the electric furnace performs wastewater treatment while injecting the wastewater containing EDTA in the range of 400°C ~ 900°C. The electric furnace is divided into the upper part heater and the lower part heater, and the lower part heat is composed of a coil-type electric heater, an inner chamber, and a basket. If wastewater treatment is performed at high temperature for a long time, it leads to equipment failure, most of which are related to the electric furnace. The electric heater is protected by a ceramic mould to protection and insulation the coil, but since the wastewater treatment process continues for a long time, the mould may break due to high temperature. In this case, as the coil-type electric heater sticks out, it comes in direct contact with the inner chamber, resulting in the electric leakage, and it may be the cause of shutting off the power of the electric furnace. For this reason, the operation of the equipment is stopped and the electric furnace is maintained during the wastewater treatment process. At this time, the wiring of the electric heater was also one of the difficulties in maintenance. In particular, the wiring of the electric heater at the lower part heater of the electric furnace may cause difficulty in replacing the inner chamber and increase maintenance time. This is the cause of the increase in the entire process period of the wastewater treatment, so the process period can be shortened through the improvement of the device.

Keywords: Chemical Cleaning, EDTA, Electric furnace, Wastewater treatment

FOSAR in the RCS Piping Performed in the SG Replacement Process

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If the defective tube increase due to the aging of the steam generator (SG) tube and the tube plugging rate of the tube increase, the SG is replaced. The replacement of the SG is a way to improve the reliability and safety of the facility for the SG. The replacement of the SG consider safety, operability, and economy. In term of safety, it is necessary to consider the increase in damage to the tube, the occurrence of a breakage of the tube, and leakage of radioactive material to the atmosphere. In the nuclear power plant while in operation, one of the ways to minimize the effect of damage to the tube of the SG is to foreign object search and retrieval (FOSAR) on the secondary side of the SG while in overhaul. In addition, in the case of SG replacement, FOSAR of the reactor coolant system (RCS) piping is performed. FOSAR of RCS piping is performed twice after cutting the SG, replacing the SG, and completing welding.

For the FOSAR of RCS piping, a robot with a camera and foreign object retrieval tool are used. The inspection method is to check the resolution of the camera and install a robot in sequence in the hot leg or cold leg 1 and 2 between the SG and the reactor to be replaced. The part where the lower part casing of the reactor coolant pump is visible while recording by rotating the camera of the installed robot. The inspection is performed on the section that can be inspected up to. If a foreign object is found, the kind, location, and special details of the foreign object including photograph are recorded and foreign object retrieval.

Since FOSAR of RCS piping is performed in high radiation areas, sufficient mock-up training on installation, operation, and removal of foreign object of the robot before inspection is conducted. Through mock-up training, the inspection time should be shortened and the exposure of workers should be minimized.

Keywords: FOSAR, Steam Generator, RCS piping, Mock-up training

Fundamental Approach to Emergency Planning Zone Analysis for Small Modular Reactor Accidents in Marine Environment

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Emergency planning zone (EPZ) for domestic nuclear power plants is subject to International Atomic Energy Agency (IAEA) standards, and there are no standards for Small Modular Reactor (SMR). Further research is also needed as there is no way to assess EPZ for nuclear accidents in the marine environment of ships propelled by nuclear power. The emergency planning zone is divided into the precautional action zone (PAZ) and the urgent protection action planning zone (UPZ). The zones around the plant prevent the severe deterministic effect and stochastic effect within PAZ and UPZ. Evaluation of EPZ requires radiation source terms, release characteristics, meteorological conditions, and protective actions. In marine environments, there are differences in weather, geographic environments, and protective actions, unlike those evaluated on the ground.

By releasing radionuclide from the reactor accident, external and internal exposure doses by cloud shine and intake are assessed by calculating radioactivity concentrations in the air based on the Gaussian plume model. The average wind speed, atmospheric stability, and radionuclide released from reactor containment are factors to evaluate the Gaussian plume model. At the same time, the external exposure is further evaluated by considering ground shine by sea surface deposition, not cloud shine. To evaluate the sea surface deposition, dry and wet deposition including resuspension should be considered the removal mechanism of the radioactive particles into the surface. The ground shine by deposition of radioactive particles is evaluated by considering a combination of the mechanism deposited on the sea surface by the gravity of radioactive particles in the atmosphere and rain wash or moisture in the air.

IAEA protective actions are used to calculate EPZ size considering house and building sheltering and taking Iodine thyroid blocking (ITB) agent. ITB agent or additional protective action should be considered since house or building sheltering is not possible in marine environments. The EPZ range will be established by applying protective action that can be considered in the dose value derived through dispersion and deposition of released radioactive materials.

Keywords: Emergency planning zone, Marine environment, Small Modular Reactor, Dispersion, Protective action

Methodological Review of Emergency Action Level Criteria for a Research Reactor

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The IAEA (International Atomic Energy Agency) and the U.S. Nuclear Regulatory Commission (U.S.NRC) have also issued guidance documents for Emergency Action Levels (EALs) for research reactors to help nuclear operators set up a nuclear emergency preparedness. Meanwhile, as Korea focuses on the criteria for issuing nuclear emergency to NPPs. its validity has always been questioned when applied to research reactors. Therefore, it is necessary to research and develop the guidance for research reactors in Korea.

In this study, cases related to domestic and international emergency action levels were presented and reviewed. In addition, it compares the characteristics and accidents of nuclear power plants and research reactors, and establish criteria of the EAL for research reactors.

It was presented in accordance with the concept of establishing EALs for research reactors, such as spatial radiation dose rates at site boundaries, monitoring radioactive material outflows and classification by emergency entry conditions, as presented by the IAEA and the U.S. NRC. Furthermore, KAERI can be used for upgrading EALs for research reactors in the future.

Keywords: Emergency Action Level (EAL), Research Reactor, Nuclear Emergency Preparedness, Emergency Planning

7분과

방사화학 (Oral)

LA-ICP-MS Analysis for Characterization of Nuclear Materials

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Laser Ablation (LA) method offers the simple sample preparation without a destructive procedure. Laser just ablates the sample surface and ablated particles with carrier gas move through sample line to the analyzer. LA combined with inductively coupled plasma mass spectrometry (ICP-MS) provides quick measurement for the determination of trace elements and isotope ratios. Without LA, pre-treatment process such as cutting and dissolving the solution sample to chemical solvents should be required.

LA-ICP-MS has been applied to analyze trace elements and isotope ratios of nuclear materials. For example, the isotope ratios of U-10 Mo fuel and irradiated nuclear fuel were successfully determined from LA-ICP-MS analysis. There analyses show that the results from LA-ICP-MS are as accurate as that obtained from ICP-MS with a destructive pre-treatment process of sample. The analysis of nuclear materials using LA-ICP-MS has a great advantage in that it can drastically reduce the amount of waste generated in pre-treatment process of solid sample. Recently, it was confirmed that the sensitivity was improved by about 100 times by applying the desolvator to LA-ICP-MS in the analysis of solid samples. This shows that analyzing nuclear materials with a smaller amount is possible, and suggests an analysis attempt that is safer and generates much smaller amount of analytical waste.

In this study, the present case of analysis of nuclear materials using LA-ICP-MS and the development status of sensitivity improvement technology of LA-ICP-MS using desolvator are to be presented.

Acknowledgements

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Keywords: Laser ablation, ICP-MS, Nuclear material, Analysis

Development of Monitoring System for Molten Salts by Laser-induced Breakdown Spectroscopy

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In the nuclear industry, the demand of molten salt matrices system increase for future usage including molten salt reactor (MSR) and pyro-chemical process. Both use high-temperature molten chloride/fluoride at 500 ~ 750°C for high thermal efficiency with 1 atm. The main challenges for monitoring this system are a detection system has to operate under high-temperature and strong radioactivity environments. The significant breakthrough in the monitoring system is the ability of stand-off detection for avoiding malfunction by high radioactivity. In addition, the sample preparation process should be neglected. This is because the nuclides in the molten salts migrate during the cooling of the molten salts. This disturbs the proper measurement of target nuclides.

For this reason, several research teams including the University of Michigan, University of Sheffield, Idaho national laboratory, Pacific Northwest national laboratory, etc. are considering laser-induced breakdown spectroscopy (LIBS) as proper technology for monitoring molten salts system. This technology has the ability of stand-off detection and unneedful sample pre-treatment process. Even though the LIBS has an optimum condition for applying to molten salt monitoring, there are some issues. Most research of the LIBS for molten salts shows the high relative standard deviation (RSD). The high value of RSD can represent the low precision. However, these issues are originated from inappropriate methodologies and insensible experimental setup.

In this research, we developed a bespoke glovebox to test this technology appropriately. This glovebox includes an optical breadboard, gas protective layer, and movable furnace. These support the proper evaluation of the LIBS for the molten salt monitoring system. The optical breadboard can make sensible control of the laser system. The gas protective layer can improve the RSD by blowing splashed sample away from the laser optics. The movable furnace can control the lens-to-sample distance consistently. This glovebox system can control the temperature up to 800°C. In the test experiment, we got acceptable RSD of LIBS spectra successfully. This value is improved the RSD by at least a factor of 10 when comparing previous research.

Keywords: Laser-induced breakdown spectroscopy, Molten salt reactor, Process monitoring, Pyro-chemical process, High precision

Synthesis of CeO_2 Nanoparticles by Using Hydrothermal Method

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Ceria (CeO_2) is widely used in heterogeneous catalysis, solid oxide fuel cells, gas sensors, UV filters, optics, and polishing materials. In recent years, CeO_2 nanoparticles have received tremendous attention because of their unique properties resulting from a low dimensionality, high surface area, and potential application as a chemical surrogate for the tetravalent actinides. In this work, nano-crystalline particles of CeO_2 have been synthesized by using the precipitation-hydrothermal method. For the sample preparation, Cerium(III) nitrate hexahydrate ($Ce(NO_3)_3.6H_2O$) as precursor and ammonia solution ($NH_3.H_2O$) as precipitator were employed as initial material. Cerium hydroxide ($Ce(OH)_3$) solution was prepared by adding 28% $NH_3{\cdot}H_2O$ with 0.5 M $Ce(NO_3)_3$ under vigorous stirring condition. The $Ce(OH)_3$ precipitate was then separated by centrifugation, washed with water, and dispersed in the water. The colloidal suspensions of $Ce(OH)_3$ were heated with PTFE-lined autoclave at 180°C for 24 hours where the $Ce(OH)_3$ precipitate transformed into a highly crystalline CeO_2 nanoparticle. The synthesized nanoparticles were characterized by using SEM-EDX and XRD. The EDX analysis confirmed that the prepared nanoparticle consists of Ce and O only, without any impurities. The SEM analysis showed that the as-synthesized CeO_2 nanoparticles grew in a spherical and octahedral shape with an average particle size of *ca.* 14 nm. The methodology employed in the present work offers a simple and versatile route for the preparation of pure CeO_2 nanoparticles.

Keywords: CeO_2, Nanoparticle, Hydrothermal Method, SEM-EDX

Characterization of Phosphorous-Functionalized Silica Microspheres

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Bis (2-ethylhexyl)phosphoric acid (HDEHP) is a conventional extractant extensively used to recover uranium by solvent extraction. Among the extraction techniques using HDEHP, solid phase extraction with porous materials is preferred to the liquid-liquid extraction due to efficiency, simplicity, and environment friendliness. In this regard, many studies have been in progress to develop various type of solid supports impregnated with phosphorous-functional materials as an effective separation agent for some actinides and lanthanides. We previously reported the stable and uniform silica microspheres with anomalous large mesopores via a surfactant-templated synthesis using n-dodecylamine as a structure-directing agent as well as HDEHP. In our present study, instead of calcination at 823 K, hydrothermal treatment for as-synthesized silica-HDEHP/n-dodecylamine microsphere composite were conducted at 343 K – 403 K. Characterization of our as-synthesized phosphorous-functionalized silica for capturing uranium in waste water was performed with scanning electron microscopy (SEM), energy-dispersive spectrometer (EDS), Fourier transform infrared spectroscopy (FT-IR), and thermo-gravimetric analysis (TGA). SEM image showed that the uniform mesoporous silica spheres were synthesized successfully with an average particle size of about 100 μm in diameter. Elemental mapping analysis using EDS showed a uniform distribution of phosphorous on the surface of large mesopores. FT-IR bands observed at 1230 and 870 cm^{-1} related to ν(P=O) and ν(P-O), respectively, indicate the presence of the intact phosphate-functional group bound with silica surface even after hydrothermal treatment. TGA showed the mass contents of total organic materials such as HDEHP and n-dodecylamine. Weight fraction of n-dodecylamine was ~ 47 wt% whereas that of HDEHP was around 8 wt% calculated based on the molar fraction of two organic materials in the initial reaction mixture.

Keywords: Mesoporous silica microsphere, Surfactant-templated synthesis, Phosphorous functionalization, Characterization, Uranium, Waste water

Solubility Measurement of PuO_2 in Natural Waters

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For the safety assessments of the underground disposal for spent nuclear fuel, the solubility data of plutonium are required in various natural water conditions, such as oxic and anoxic groundwaters and pore waters in the Engineered Barrier System (EBS) under disposal conditions. It is well known that plutonium chemistry in aqueous solutions is very complicate due to its redox sensitivity. Plutonium hydrous oxides could be dissolved from 10^{-11} to 10^{-3} M in neutral pH depending on oxidation state of plutonium. Solubility products of plutonium solid phases formed with CO_3^{2-}, SO_4^{2-}, PO_4^{3-}, Cl^-, etc. are scarce in literatures. We can only estimate solubility of plutonium in the underground disposal environments by modeling with large uncertainty.

In the present work, solubility of PuO_2 in natural water (groundwater, rain water reacted with concrete and sea water) are investigated under aerobic conditions. Experimental difficulty comes mainly from chemical amounts of plutonium that we can handle in a laboratory. Characterization of plutonium solid phase before and after solubility measurement, phase separation for accurately measuring concentrations of dissolved plutonium, speciation of dissolved plutonium (oxidation state and complexation) could not be carried out in this work. One mg of PuO_2 was added into the natural water of 30 mL and plutonium concentrations in supernatant were measured by liquid scintillation counter (TriCarb4910TR, PerkinElmer) as a function of reaction time. Dissolution of PuO_2 reached to equilibrium state after 2 months. The dissolved concentrations of PuO_2 in the groundwater samples were determined in between solubility of $Pu(IV)O_2$ and $Pu(VI)O_2(OH)_2$. However, it is difficult to directly identify the existence of Pu(IV) or Pu(VI) species, colloidal Pu species, any other plutonium complexes in natural water samples. Redox behaviors, complexation, solidification and chemical speciation of plutonium in underground disposal environments have to be continuously investigated in order to support the spent nuclear fuel disposal.

Keywords: Plutonium, Solubility, Natural Water, Groundwater, Speciation

Comparison of Chemical Analysis Methods for I_2 and CH_3I Concentrations in Aqueous Samples

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The behavior of radioactive iodines has long been of interest in the nuclear research and industrial fields. I-131 is a radionuclide that determines the EAB (exclusion area boundary) of a nuclear power plant, and I-129 is a nuclide that affects the disposal period of spent nuclear fuels. In addition, the behavior of radioactive iodines in the nuclear fuel pretreatment process (eg.: dissolution or voloxidation process) or in safety facilities (eg.: containment filtered venting system) for coping with severe accidents at nuclear power plants is one of the important issues to be considered. Against this background, the analysis of the concentration of volatile iodine species, I_2 and CH_3I, is one of the essential technologies for the evaluation of the behavior of iodine. In this presentation, the accuracy was compared by measuring the concentration of I_2 using the I_2/I_3^- equilibrium relationship method and the toluene extraction method, respectively. In addition, the measurement accuracy when measuring the CH_3I concentration by GC-MS (gas chromatography-mass spectrometer) was evaluated.

Keywords: Chemical analysis, I_2 and CH_3I, Aqueous samples, GC-MS

Radiochemical Separation of Pu, Am, Tc, Sr, Nb, Fe and Ni Isotopes in Neutron Dosimeter Samples From the Nuclear Power Plant Using Extraction Chromatographic Resins

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This study presents a rapid and quantitative sequential radiochemical separation method for Pu, Am, Tc, Sr, Nb, Fe and Ni isotopes in neutron dosimeter samples from the nuclear power plant with extraction chromatographic resins. After radionuclides were leached from the neutron dosimeter samples with concentrated HCl and HNO_3, the radionuclides such as Pu, Am, Nb and Fe isotopes were coprecipitated with Fe after filtering the leaching solution with 0.45 micron HA filter, while the Sr, Tc and Ni isotopes were in the solution. Pu and Am isotopes coprecipitated with Fe were sequentially purified with anion exchange resin and TRU resin, respectively, on the other hand Tc and Sr isotopes in the solution were purified with the TEVA resin and the Sr resin, respectively. Also, Nb and Fe isotopes were separated through anion exchange resin column from Fe coprecipitation step, while Ni isotopes were purified with dimethylglyoxime (DMG) precipitation in the solution. After α source preparation for the purified Pu and Am isotopes with micro-coprecipitation method, Pu and Am isotopes were measured using alpha spectrometry. Also, Tc, Sr, Nb, Fe and Ni isotopes were measured using a low level liquid scintillation counter. The radiochemical procedure for Pu, Am, Tc, Sr, Nb, Fe and Ni isotopes investigated in this study has been applied to the samples for decommissioning main structure of nuclear power plant after method validation.

Keywords: Sequential Radiochemical Separation, Neutron Dosimeter Sample, Pu, Am, Tc, Sr, Nb, Fe and Ni Isotopes, Decommissioning

Determination of ^{32}P of RI Wastes by Solvent Extraction and Cherenkov Counting

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Radioactive Isotope (RI) waste have low-level beta-emitters with short half-lives, which is impossible to measure their inventories by a nondestructive analysis. Thus, nondestructive analysis is often required to determine the radionuclide inventory of the drums for disposal. A variety of radionuclides such as ^{3}H, ^{14}C, ^{32}P, ^{35}S, ^{90}Sr, ^{125}I, ^{147}Pm and γ-emitter (^{60}Co, ^{137}Cs, ^{99m}Tc, ^{131}I) are generated from hospital or industrial laboratory. Among them the use of phosphorous-32 (^{32}P) has greatly increased in the field of medicine, biochemistry and molecular biology and, the generation of ^{32}P in RI wastes have steadily increased. ^{32}P is a commonly used radionuclide, emitting beta particles with a maximum energy of 1710.7 keV and a half-life of 14.3 days. In this study, we separated ^{32}P from RI wastes based on the water-organic solvent extraction and determined ^{32}P by counting Cherenkov radiation. Due to its high counting efficiency, liquid scintillation counter (Tri-Carb 3180TR/SL, Perkin-Elmer) was used for detection of radioactivity. The results of procedure validation are as follows: The average recovery rate of ^{32}P was 93.91±0.03% and the counting efficiency of ^{32}P by LSC based on the Cherenkov radiation was 0.4810. We anticipate that the method developed in this work can be used for the determination of ^{32}P in radioactive waste generated from the RI utilizing institutes.

Keywords: P-32, Water-organic solvent extraction, Radionuclide, Cherenkov counting

7분과

방사화학 (Poster)

Introduction to a Method for Improving Reliability of Chemical and Radionuclide Analysis Data of Nuclear Fuel Crud Samples

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Concentration analysis of high-level radioactive materials is generally performed with the minimum amounts required for analysis due to difficulty in handling radioactive samples. For this reason, it is difficult to secure high measurement reliability that can be obtained by the multiple analysis. Meanwhile, radionuclides with different half-lives contained in neutron-irradiated materials, such as nuclear fuel crud, have a specific radioactivity concentration ratio over time. In addition, it has certain ratios between the concentrations of the radionuclides and their source element concentrations. Using these relationships, it is expected to improve the reliability of the chemical and radionuclide analysis data of the radioactive samples. In this presentation, we introduced a method to improve the reliability of analysis data by using the radioactivity concentrations of radioactive cobalts and zincs formed by neutron irradiation in nuclear fuel cruds.

Keywords: Chemical and radionuclide analysis, Nuclear fuel crud, Concentration analysis

Determination of Chelating Agents in a Variety of Radioactive Waste Using UV-Vis Spectrophotometer

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A variety of radioactive wastes are generated through decommissioning of nuclear facilities. Thus, different analytical techniques should be developed and optimized to assure safe management of the wastes. Chelating agents are generally used to treat the contaminated materials and sites while decommissioning. Chelating agents, such as EDTA, NTA, and citric acid, potentially enhance the migration of radionuclides such as ^{60}Co, ^{238}Pu, ^{241}Am, and ^{90}Sr away from disposal sites. Therefore, a precise analysis for the contents of these chelators in radioactive wastes is required for safe operation of the disposition site. However, only few method is available to analyze the contents of the chelators presenting in a variety of matrix, such as concrete, metals, plastics, vinyl and rubbers.

In this study, the method for determination of the chelators in wastepaper, has been examined to extend the scope of analysis to various types of samples such as plastics, vinyl and rubbers in order to satisfy recent requests. In addition, the limit of detection for this method was evaluated to improve an analytic accuracy. This method consists of two parts; an extraction process and sequential measurements of absorbance or transmittance at a specific wavelength using UV-Vis spectrophotometer to determine the concentration of the chelators. The suggested method was successfully applied to radioactive plastics, rubbers, and vinyl wastes.

Keywords: Decommissioning, Chelating agents, Decontamination, Extraction, UV-Vis spectrophotometer, LOD

Synthesis of Three Dimensional Framework Structure for Immobilization of Radioactive Waste

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To immobilization of radioactive waste, among the materials, zirconolite is known as a derivative of the pyrochlore structure. Most of the commonly reported zirconolite structures, based on the formula $CaZrTi_2O_7$, are composed of two types of TiO_6 octahedra and one type of $Ti(Zr)O_5$ polyhedra, similar to the cubic pyrochlore structure in the [111] planes. In the details, however, the zirconolite structure is perfectly described, when pressure is applied to the pyrochlore structure along the [111] direction and the HTB layers are arranged at perpendicular to the c-axis direction. Based on the geometrical advantage of zirconolite framework has dramatically attracted attention as a material that immobilizes the elements as fission products, occurring in high-level nuclear reactor wastes. A novel metal oxide similar to a zirconolite with 3-D framework containing K^+, In^{3+}, Ti^{4+}, and Ti^{6+} have been synthesized, as crystals and pure polycrystalline phases through hydrothermal reaction. The metal oxide structure has been measured by single-crystal X-ray diffraction. Consist of the K-In-Te-O material crystallize in the orthorhombic space group, *Cmcm* (No.63), with a 3-D framework structure including 1-D channel consisting of Te/InO_6 octahedra, and TeO_4 polyhedra. The synthesis of product, determination of the crystal structure, and characterization will be presented.

Keywords: 3-D framework material, Zirconolite, Immobilization, Single crystal X-ray diffraction

Determination of Radioactivity of Phosphorus-32 and -33 in RI Waste Using Liquid Scintillation Counter

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The KAERI Radioactive Waste Chemical Analysis Center is conducting analysis of radionuclides in Radioisotope (RI) waste from opened sources generated by users such as domestic medical institutions, research institutions and industrial institutions. Some radionuclides in RI waste are beta-ray emitters with short half-lives, such as ^{32}P, ^{33}P, ^{35}S, ^{90}Sr, ^{125}I, ^{147}Pm, which requires a destructive analysis method for the evaluation of the nuclide inventory. In this work we studied the method of separating and measuring ^{32}P and ^{33}P in these RI wastes. Phosphorus is widely used in medical fields, genetic engineering, and biotechnology research fields. ^{32}P is a radionuclide with a half-life of 14.3 days, emitting beta particles with a maximum energy of 1710.7 keV. ^{33}P is also beta emitting particle with a half-life of 25.3 days and maximum energy of 249 keV.

Since radioactive phosphorus is a pure beta nuclide, it is common to separate chemically in RI waste and measure it using a Liquid Scintillation Counter (LSC). Despite a large energy separation of ^{32}P and ^{33}P, the LSC that measures electrons with a continuous energy spectrum hardly differentiates their spectra by setting an energy window in the device. In this work, we were able to determine the radioactivity of ^{32}P and ^{33}P by measuring both Cherenkov counting and scintillation counting on one sample. Therefore, this method is applicable to RI waste analysis.

Keywords: Liquid Scintillation Counting, Cherenkov radiation, Radioactive phosphorus, Radioactive isotope waste

Analytical Procedures for Measuring ^{35}S in Radioisotope Wastes

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The use of ^{35}S labeled compound have been increasing in the biomedical, industrial and research organizations and the need for measurement of ^{35}S in radioisotope waste is also increasing. In particular, ^{35}S has a relatively short half-life of 87 days with emitting only beta particles. In this regards, an appropriate analytical procedure is required to have high recovery yields within a short half-life.

In this study, the analytical procedures of ^{35}S are carried out in three steps; the chemical leaching step, the oxidation step and the precipitating step. As the first step, the leaching process is performed using nitric acid in the heated and closed Teflon vessel. Then, the oxidation of sulfur was carried out in the presence of the oxidants such as sodium bromate and hydrogen peroxide. In the last precipitating process, samples have been prepared for gas proportional counter (GPC) by precipitating sulfate to barium sulfate. The results of simulated experiments using the procedures demonstrated here showed high recoveries of ^{35}S.

Keywords: Radioisotope wastes, S-35, Gas proportional counter (GPC), Beta-emitter

Evaluation on Regeneration and Reuse of Sr-spec Resin

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A considerable amount of the second radioactive waste such as spent resin and liquid waste were formed during the radioanalytical determination of radionuclides. In this work, we have focused on the evaluation of reusing the spent Sr resin. After ^{90}Sr was separated from matrix elements and other regulatory radionuclides such as Ni and Fe by co-precipitate with CaC_2O_4, Sr-spec resin was used for ^{90}Sr separation from Ca. The used resin was regenerated using 0.05 M and 8 M nitric acid, which removed remained calcium and matrix elements as well as strontium from the resin. The performance of the resin was assessed regarding ^{90}Sr recovery, column capacity, and cross-contamination. The recovery of ^{90}Sr was measured by ICP-OES (Inductively Coupled Plasma Optical Emission Spectrometry). To examine the cross-contamination between samples, radioactivity of ^{90}Sr was detected by LSC (Liquid scintillation Counter). The values before and after the regeneration were compared to evaluate whether the Sr-spec resin could use repeatedly and thus the regeneration could reduce the second waste produced from the radiochemical analysis of radioactive waste.

Keywords: ^{90}Sr separation, Sr resin, Regeneration, Radioactive wastes

Development of Multi-Analytical Method for Particulate Matter of Environmental Samples Using Nuclear Track Technique

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A unique sample preparation method was developed for the detection and analysis of particle containing nuclear materials in environmental samples. The basic sample assembly consisting of a particle layer made of recovered particles and a dedicated tool can be used in common regardless of particle detection method and analysis method. In the case of particle detection using the fission track (FT) technique, the target particle can be detected from the fission tracks formed on the FT detector by stacking the FT detector with the particle layer and irradiation. By stacking an alpha ray detector on the basic sample assembly, alpha ray emitting particles such as plutonium can be detected. These two detection methods can be applied sequentially to same particle layer by using a dedicated tool. In addition, the square dedicated tool designed in this study enables simple and accurate detection of the target particles from the track without any fiducial point marking. Since the particles detected can be displayed simply with a laser, they can be applied to various analysis methods such as LA-ICP-MS as well as TIMS depending on the purpose without any pretreatment.

Keywords: Particulate matter, Isotopic composition, Nuclear track, Environmental samples

Structural Determination and Characterization of Metal Oxides for Immobilization of Radioactive Waste

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In terms of immobilizing high-level radionuclides, titanium-based materials have been extensively used. In this regard, we previously considered the applicability of Titanium Tellurium Oxide, $TiTe_3O_8$ for a solid waste form. Ti^{4+} can be exchanged from various multivalent cations (Er^{3+}, Sn^{4+}, Ce^{4+}) which retained in the same octahedra structure at the Ti^{4+} site due to its structural effects. In this paper, we discovered the site can contain hexavalent tellurium cation, in form of $Ti_{1-x}Te_xTe_3O_{8+x}$ (x = 0, 0.1, and 0.12). The crystal structures of the materials were confirmed and refined using single-crystal X-ray diffraction. $Ti_{1-x}Te_xTe_3O_8$ (x = 0, 0.1, and 0.12) materials have three-dimensional framework structures consisting of TiO_6 or Ti/TeO_6 octahedra and asymmetric TeO_4 polyhedral groups. Detailed characterizations of the prepared materials, including Raman spectra and thermogravimetric analysis were performed. In addition, we observed that the material framework remained unchanged under gamma-ray irradiation with 40 kGy dose by powder X-ray diffraction to ensure that the structure can be kept stable by radiation from radioactive nuclides.

Keywords: Solid waste form, Mixed-valent titanium tellurium oxides, Structural characterization, Gamma-ray irradiation;

Re-evaluation on Radioactivity of Irradiated Fission Track Detector

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Fission track detector is a special method to detect fissile material in environmental samples. Particles of fissile material with extremely low amount in the samples may not be collected by manual treatment. Fissile material deposited on fission track detector undergoes nuclear fission induced by neutron flux in a research reactor. Radiation induced by nuclear fission damages glass or polymer of the detector, and the damaged trace is regarded as fission track. The particles can be located in the center of multiple fission track on the detector.

A sample container and buffer material along the fission detector is activated by thermal neutron flux. From the previous study, it has been found that their radioactivity of impurities activated is greater than the radioactivity of fission products from fissile material, and they are main concerns for requirement of NSSC Notice on exemption criteria. For more accurate calculation of radioactivity from the activated impurities, SCALE/ORIGEN-6.2 was used for computational modelling while ORIGEN-2.2 was used in the previous study. The major difference for applying SCALE/ORIGEN-6.2 is that multi-group neutron flux can be applied rather than one group flux for ORIGEN-2.2. The multi-group neutron flux was obtained from High-Flux Advanced Neutron Application Reactor (HANARO), and it was converted to a certain format for SCALE. The elemental composition of each material including the impurities from polycarbonate detector and polyethylene container is same as the input of previous study. For fissile material, the natural abundance of uranium was used and artificial abundance was used for plutonium.

The newly evaluated radioactivity was compared to previous study. However, it still meets the requirement on exemption criteria regarding radiation safety, with several days of cooling after irradiation in HANARO.

Keywords: Exemption, HANARO, Fission track, Environmental samples

A Semi-mechanistic Model for UO_2 Pellet Oxidation at 400, 500 and 700 ^oC

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In the fabrication of dry recycling nuclear fuel pellets, fission product such as tritium, cesium, krypton and xenon are partially released during voloxidation process where UO_2 pellet is transformed to U_3O_8 powder and completely released at high temperature treatment. Although several fission product release models were developed, these model are not applicable to the description of fission release behavior in dry recycling process since the models do not account for oxidation of UO_2 pellet during thermal treatment under oxidative environment. For the description of fission product release behavior of dry recycling process, we preferentially developed a semi-mechanistic model for UO_2 pellet oxidation applicable to various shape and oxidation condition.

For model development, the following assumption are made.

- All pellets are approximated to sphere having same ratio of surface to volume
- Cracking occurs due to the formation of U_3O_8 occurs in solid surface yielding to the increased reactive surface area
- Porosity of pellet changes as reaction proceeds
- Oxygen diffusion follows 1D spherical diffusion
- Sphere is equally divided into symmetrical shell

The factors affecting oxidation of pellet, oxidation temperature, size of pellet (diameter and height), ratio of diameter to height, oxygen concentration, are also considered in model development.

Since mass balances of oxygen molecule and solid reactant are described by partial differential equations, for solving equations, finite volume method (FVM) or method of line (MOL) converting partial differential equation to sets of ordinary differential equations were applied, yielding to same calculation results. For given six systems, the developed model was found to describe conversion-time behavior with temperature-dependent three parameters optimized by regressing conversion-time data of different pellets size and temperature ranging from 400^{o}C to 700^{o}C. It was found that, except for one system, the developed model closely reproduced experimental data and predicted other system well. Since the present model is expected to be applicable to UO_2 oxidation behavior of various pellets, we are planning to incorporate the developed model into 1D based two-stage spherical diffusion model for the prediction of fission product release behavior of U_3O_8 powder.

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Keywords: Diffusion model, Solid-gas reaction, UO_2 oxidation, Cracking, One-dimensional

Quantification Analysis of Various Iodine Species by Using Colorimetry

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In the context of high-level waste disposal, iodine is considered to be one of the major radionuclides owing to its high solubility along with low sorption affinity to natural mineral surfaces. For the precise characterization of the sorption behavior of iodine, reliable analysis methods to quantify the iodine concentration in aqueous solution are requested. The objective of the present work is to analyze the aqueous concentration of various iodine species such as iodine (I_2), iodide (I^-), tri-iodide (I_3^-), and iodine-starch by using UV/VIS absorption spectroscopy.

Iodine and iodide stock solutions were prepared by congruent dissolution of solid iodine (Sigma-Aldrich) and sodium iodide (Sigma-Aldrich), respectively. Total iodine concentration in the sample solution was adjusted in the range from 5×10^{-6} M to 1×10^{-4} M. The sample containing tri-iodide species was synthesized by adding iodine solution of $[I_2] = 1.1\times10^{-6}$ M – 2.1×10^{-5} M to sodium iodide solution with $[NaI] = 8.4\times10^{-2}$ M. For the preparation of iodine-starch sample, 4 ml of 0.5 M H_2SO_4 (Sigma-Aldrich), 1 ml of 0.1 M H_2O_2 (30%, Sigma-Aldrich), and 1 ml of 1 g/100 ml starch solution were added to the 4 ml of $[NaI] = 2.0\times10^{-5}$ M – 3.0×10^{-4} M solution. The ionic strengths of the samples were set to be 0.01 M to 1 M with NaCl (Sigma-Aldrich) and $NaClO_4$ (Sigma-Aldrich).

The result obtained by using UV/VIS absorption spectroscopy revealed two characteristic absorption peaks of 204.5 nm ($\varepsilon = 1.94\times10^4$ l/mol·cm) and 460.0 nm ($\varepsilon = 6.83\times10^2$ l/mol·cm) for iodine solution. However, relatively blue-shifted absorption peaks of 194.5 nm and 226.0 nm for iodide solution were observed with somewhat decreased molar absorptivity of $\varepsilon = 1.44\times10^4$ l/mol·cm at 194.5 nm and $\varepsilon = 1.36\times10^4$ l/mol·cm at 226.5 nm compared with iodine sample. In addition, tri-iodide species showed relatively red-shifted peaks located at 288.0 nm and 351.0 nm with significantly increased molar absorptivity values. The molar absorptivity of the iodine-starch complex observed at 607.5 nm was remarkably influenced by the reaction time for the complex formation. In the present work, 30 – 40 minutes of the reaction time indicated the best result in terms of the molar absorptivity.

In the present work, the spectroscopic properties of various iodine species were determined in terms of UV/VIS absorption spectra and molar absorptivity values. The obtained result can be further employed for the precise quantification of iodine species in aqueous solution together with the characterization of sorption behavior of iodine species onto mineral surfaces.

Keyword: UV/VIS absorption spectroscopy, Iodine, Iodide, Tri-iodide, Iodine-starch

Sample Pretreatment for Determination of Ce, Co, Cs, Fe, Nb, Ni, Re, Sr and U in Cemented Ion Exchange Resin Using High-Performance Microwave Digestion System

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A sample pretreatment procedure was developed to measure radioactivity in cemented ion exchange resin samples. The cement plus ion exchange resin adsorbing 9 elements (Ce, Co, Cs, Fe, Nb, Ni, Re, Sr and U) was heated on a hot plate and a high-performance microwave digestion system (HP-MDS) in the presence of HNO_3 (10 mL)-HCl (1 mL)-HF (0.25 mL)-H_2O_2 (0.5 mLx6). To measure the recovery of Fe and Sr desorbed from ion exchange resin, we need to know the Fe and Sr content present in the cement. Calcium oxide (CaO) and silicon dioxide (SiO_2) are the main component of cement. SiO_2 is dissolved only in the presence of HF, but calcium as a CaF_2 has a low solubility product constant. The precipitation of CaF_2 is formed when excess HF coexists with calcium ions. Thus, the cement was dissolved in two steps. First, cement, HNO_3 and HCl were added to the vessel and the vessel was heated on a hot plate and an HP-MDS. After centrifugation, most of the insoluble residues of silicon dioxide were completely dissolved with HF-HNO_3-HCl without heating. The amount of elements in the each obtained solution was measured using ICP-AES and ICP-MS. Seven elements except Cs and Nb were quantitatively recovered. However, the recovery of Cs and Nb was less than 95% regardless of the volume change of HNO_3-HCl-HF-H_2O_2. Thus, Cs and Nb remaining in ion exchange resin were desorbed once more operating an HP-MDS following filtration of insoluble residues. Finally, nine elements in simulated solidified ion exchange resin were quantitatively recovered with a mixture of HNO_3-H_2O_2-HCl-HF using a hot plate and an HP-MDS. This pretreatment procedure will be applied to measure radioactivity in cemented ion exchange resin samples generated before 2002.

Keywords: Pretreatment, Determination, Cement, Ion exchange resin, Microwave digestion

Performance Testing Result of Distillation Kit for Tritium Measurement

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Tritium is a pure beta-particle emitting radionuclide with low beta energy (maximum energy: 18.6 keV) and it is readily interfered with other co-existing radionuclides meaning that its quantification is tough. So, a radiochemical purification is needed to prevent the interference prior to counting tritium by liquid scintillation counter. Distillation technique is a representative test method for purification tritium as HTO such as ISO 9698 and ASTM D4107 etc. Our Distillation Apparatus for a simultaneous assay of carbon-14 and tritium in radioactive wastes is well-suitable to purify tritium; however, that has a limit of time-consuming like a considerate observation, and labor-intensive work such as complicate washing. Distillation Kit, which is developed by Prof. Philip Warwick's research group at University of Southampton, is quite more efficient than our distillation apparatus in some points. It does not need a considerate observation during distillation, and washing the distillation kit is easier.

Along with the merit of the above mentioned, it showed that the blank count rate was manageable and there was no memory effect observed. It showed that recovery of tritium radioactivity was satisfying according to the standard of ISO 9698, which is stated that the radioactivity recovery is 95%. To check the cross-contamination and the effect of the position of distillation beaker, the final performance testing was conducted. It was confirmed that there was no cross-contamination observed in low concentration of tritium, which was surrounded with medium- and high-concentration of tritium. It was also confirmed that the position of distillation beaker had little effect on the recovery of tritium radioactivity. This series of processes were conducted to validate the distillation kit applied with the ISO 9698 for fulfillment of ISO 17025, "General requirements for the competence of testing and calibration laboratories". The more detailed results will be presented at the conference.

Keywords: Tritium, Distillation, ISO 9698, ISO 17025, Performance Testing

Preliminary Study on the Synthesis and Properties of Radiation-Resistant Metal Organic Framework for N_2, CO_2, Kr and Xe Selective Sorption

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The low concentration of Kr and Xe in the atmosphere and using cryogenic installations to isolate these gases greatly complicates the process and increases energy costs. Recently, the metal-organic framework (MOF) has become widespread and is successfully used for the separation, adsorption and storage of gaseous radioactive materials. The ability of MOFs to adsorb gases at room temperature can significantly improve the cost-effectiveness of the gas purification process. Radiation-resistant metal-organic framework SIFSIX-3-Cu is a periodic 4^4 square grids pillared by silicon hexafluoride anions, which has been already reported as a promising adsorbent which successfully allows the selective adsorption of CO_2, N_2, Kr and Xe under highly-radioactive environments.

This work describes synthesis and characterization of the SIFSIX-3-Cu based on pyrazine/copper(II) hexafluorosilicate. The composition of synthesized MOF was confirmed by XRD (X-ray Diffraction) for comparison with CCDC database. The CO_2, N_2, Kr and Xe adsorption and desorption isotherm measurements were performed with a Micromeritics ASAP 2020 Plus 2.00 at 77 K, 273 K, 298 K and 298 K respectively. Samples were activated under vacuum at room temperature for 12 hours and at 323 K for another 12 hours. Pore size and pore volume distributions were calculated using the DFT (Density Functional Theory) method from CO_2 adsorption isotherm at 273K. Comparative characteristics are also given using TGA (*Thermogravimetric Analysis*) before and after dehydration at 180ºC for 3 hours, BET surface area and CO_2, Kr and Xe adsorption capacities.

The identical BET surface area 188 m^2g^{-1} was determined from the CO_2 adsorption isotherm at 273 K and 298 K, calculated pore size for the SIFSIX-3-Cu equals 3.23 Å. The TGA analysis of the SIFSIX-3-Cu observed a weight loss of about 10% for the both samples (before and after drying at 180ºC) in the range of 50–150 ºC attributed to water molecule and a gradual loss was observed above 200ºC and attributed to the organic decomposition of MOF.

Keywords: SIFSIX-3-Cu, Metal organic framework, Radiation – resistant, Gas sorption

Sequential Determination of Americium, Curium, Plutonium and Strontium-90 in Sludge Samples for Nuclear Power Plant

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To meet the acceptance criteria for disposal of low and intermediate level radioactive waste (LILW), 95 percent of the total radioactivity are determined and the concentration of the following 14 radionuclides must be determined. The corresponding nuclides are ^{3}H, ^{14}C, ^{55}Fe, ^{58}Co, ^{60}Co, $^{59/60}Ni$, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I, ^{137}Cs, ^{144}Ce and Gross alpha activity. Gross alpha activity is the sum of alpha particles emitted by all alpha emitting nuclides. This means that there is no need to determine the concentration of individual alpha emitting Nuclides (IAENs). However, from 2021, if the gross alpha activity exceeds 10 Bq/g, IAENs analysis results must be determined for each sample. It depends on the source of radioactive wastes; KHNP need to identify $^{238/239/240}Pu$, $^{242/244}Cm$, and ^{241}Am and KAERI additionally needs to analyze $^{234/235/237}U$. Therefore, it is essential to develop and validate analytical method for IAENs in radioactive waste.

In this study, a sequential analytical method using TEVA-DGA-Sr resin was established to determine IAENs and ^{90}Sr, one of regulatory nuclide. Sludge was selected as target sample matrix, one of the major waste generated from the nuclear power plant. Instead of real sludge sample from nuclear power plant, we evaluated the developed method using the industrial sludge certified reference material (NIST SRM 2782). The samples were totally decomposed using microwave digestion. After that, actinides and Sr were coprecipitated using CaF_2-LaF_3. The precipitate was dissolved in 8 M HNO_3−0.25 M H_3BO_3 solution and then followed extraction chromatographic separation using TEVA-DGA-Sr resin. ^{241}Am, ^{244}Cm, ^{242}Pu and ^{90}Sr (stable Sr) used as tracer were determined by alpha spectrometer and ICP-OES. The average recovery of ^{241}Am, ^{244}Cm, ^{242}Pu and Sr were 111 ± 2%, 105 ± 3%, 97 ± 3% and 91 ± 3%, respectively. Based on the results of analytical recovery and measurement performance, developed method could be used for rapid determination of IAENs and Sr in radioactive wastes. The residual mass of La and Ca in the final elution solution was less than 3 ug, of which effects are negligible on the alpha spectroscopic analysis. All alpha spectra have a good peak resolution of under 28 keV (FWHM). Developed method can be used for rapid determination of radioactive wastes. Furthermore, it is expected that laboratory radioactive waste, which can increase the cost of analysis, will also be reduced through the sequential separation.

Keywords: Actinide, Strontium, Radioactive waste, Sequential separation

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