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Contents

1분과 핵주기정책 · 규제 및 비확산 [ORAL]

- 1-O-01 Development of Research, Education, and Training Program on Nuclear Fuel Cycle and Economics Towards Advanced Reactor Technology at Virginia Commonwealth University 41
Supathorn Phongikaroon (Virginia Commonwealth Univ.)
- 1-O-02 Tracking Footprints of Nuclear Security Centre of Excellence in the Republic of Korea 42
Jae-Jun Han, EunBee Park, Byung-Woo Shin, DongHoon Shin (KINAC)
- 1-O-03 Investigating Improvement Strategies for Nuclear Nonproliferation and Security Awareness Programs in Domestic Human Resource Development 43
Junghyun Na (KINAC)
- 1-O-04 Language Matters: A Corpus-based Analysis of the Korean Translation of ‘(High-Level) Radioactive Waste’ in Mainstream Korean News Media 44
Kyung Hye Kim (Dongguk Univ.), Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)
- 1-O-05 Consideration of Nuclear Nonproliferation Technology Development Strategies for Advanced Nuclear Fuel Cycle 45
Seong-Kyu Ahn, Young-Eun Jung, Jae Soo Ryu (KAERI)
- 1-O-06 Analysis of the Nuclear Materials Attractiveness Through New FOM Including the Nuclear Material Quantification Factor 46
Danwoo Ko, Seung Min Woo (Kyung Hee Univ.), Yonhong Jeong (KINAC)

1분과 핵주기정책 · 규제 및 비확산 [POSTER]

- 1-P-01 Implementation of a Nuclear Regulatory Law Search System Using the RAG (Retrieval Augmented Generation) Model 49
Sungho Yoon (KINAC)
- 1-P-02 Gap Analysis of Regulatory Standards for Applying Risk-Informed and Performance -based Regulation in Nuclear Cybersecurity of Republic of Korea 50
Hyunjoo Lee, Subong Lee, Taehee Kim (KINAC)

1-P-03	Improvements in Nuclear Export and Import Control System (NEPS) Functionality and Future Enhancement Directions	51
	Siwon Kim, Hee Su Choe (KINAC)	
1-P-04	Analysis on the Operation of the Nuclear Pre-Consultation System	52
	Hyun Doo Kim, Hee Su Choe (KINAC)	
1-P-05	Review of the Necessity for a Self-Inspection System in Post-Management Obligations of Nuclear Power Plant Technology Export License	53
	Hyun Doo Kim, Su-Hyeon Kim (KINAC)	
1-P-06	Regulatory Improvements for Safeguards Implementation From the Design Phase	54
	Chul Heo, Seungmin Lee (KINAC)	
1-P-07	Comparison of Policies During Trump and Biden Administration: Focusing on Nuclear Nonproliferation Issues	55
	Hojung Do, Yonhong Jeong, Dongjin Kim (KINAC)	
1-P-08	Analysis of SMR Export Business Status in Korea and Improvements for SMR Export Control	56
	Su-Hyeon Kim, Hyun Doo Kim (KINAC)	
1-P-09	A Study on How to Improve Nuclear Export Control Outreach Program	57
	Hee Su Choe, Siwon Kim (KINAC)	
1-P-10	Research on Security Regulatory Documentation Between the Domestic and NuScale Case	58
	Subong Lee, Hyunjoo Lee, Taehee Kim (KINAC)	
1-P-11	FOM-based Nuclear Proliferation Assessment on the Application of HALEU in ATF	59
	Jee Min Han, Seung Min Woo (Kyung Hee Univ.)	
1-P-12	Measurement of LiCl-KCl Salt on the Dipstick Using FO LIBS and Spectral- based Normalization Methods for the Quantitative Analysis	60
	Se-Hwan Park, Byung Suk Park, Seong-Kyu Ahn, Jonghui Han (KAERI)	
1-P-13	Feasibility Study on Random Number-based MBE Method for MUF Uncertainty in Benchmark Bulk Handling Facility	61
	Hyun Cheol Lee, Jung Youn Choi, Hyun Ju Kim, Yewon Kim, Haneol Lee (KINAC)	
1-P-14	A Study on the Improvement and Optimization of KAERI Export and Import Control System	62
	Seong-mi Han, Hyun-Jo Kim, Mun-Young Ryu, Ho-Jin Hwang (KAERI)	

1-P-15	Considerations for Adopting the Obligation Code for Domestic Management of Imported Internationally Controlled Materials	63
	Hee Su Choe, Si-won Kim (KINAC)	
1-P-16	Development of Nuclear Proliferation Risk Assessment Methodology Based on Data Analysis and Deep-Learning Technologies Using Open-Source Information	64
	Jae Min Lee, Soo Hwan Lee, Saerom Son, Eun-ha Kwon (MCG)	
1-P-17	A Study on the Requirements and Design for Development of Decision Support System for Nuclear Export Control	65
	Mun-Young Ryu, Hyun-Jo Kim, Seong-mi Han, Ho-Jin Hwang (KAERI)	
1-P-18	A Study on the Revision of the Regulation on the Export and Import Approval for Nuclear Materials to Improve the Nuclear Nonproliferation System	66
	Hee Su Choe, Hyun Doo Kim (KINAC)	
1-P-19	Proposal of the Nuclear Material Accounting Method for Molten Salt Reactors ..	67
	Ji Hoon Lee, Seung Min Woo (Kyung Hee Univ.)	
1-P-20	Quality Management System for KINAC's R&D Processes Based on ISO9001: 2015	68
	Hyun Young Kim, Jaeyeong Jang (KINAC)	
1-P-21	Developing R&D Outcomes Management System for Effective Knowledge Administration of the Organization	69
	Hyun Young Kim, Jaeyeong Jang (KINAC)	
1-P-22	Economic Feasibility Analysis for Nuclear Weapons Programs	70
	Yonhong Jeong, Dongjin Kim (KINAC)	
1-P-23	DPRK's ELWR Analysis Based on VVER Light Water Reactor Fuel	71
	Dongjin Kim, Hojung Do, Yonhong Jeong (KINAC)	
1-P-24	Consideration of Artificial Intelligence Technology's Impact on Nuclear Non-Proliferation	72
	Yonhong Jeong, Dongjin Kim (KINAC)	
1-P-25	Current Status Review of AUKUS: Concerns and Responses for Nuclear Nonproliferation	73
	Yuji Jeong, Hojung Do, Hana Seo (KINAC)	
1-P-26	Analysis of the Nuclear Material Accountancy System for Bulk Handling Facilities	74
	Ju-Ang Jung, Byung-hee Won, Jinha Choi, Hyun-Jo Kim (KAERI)	

1-P-27	Development of Cybersecurity Training Equipment Based on Nuclear Power Plant Control and Instrumentation Systems	75
	Junghyun Na (KINAC)	
1-P-28	Analysis of the Status of Findings Issued in NMAC Inspections and Suggestions for Improvement Directions	76
	Chul Heo (KINAC)	
1-P-29	Components of Cybersecurity Incident Response Records	77
	Taehee Kim, Hyunjoo Lee, Subong Lee (KINAC)	
1-P-30	Simulation Study to Apply a Compton Suppression Technique on CZT-based Spent Nuclear Fuel Characterization System	78
	Wooseong Hong, Geehyun Kim (Seoul Natl. Univ.)	

2분과 사용후핵연료 처분전관리 [ORAL]

2-O-01	Laser Induced Breakdown Spectroscopy (LIBS) and Its Success in Used Nuclear Fuel Disposal and Reprocessing Technology	81
	Supathorn Phongikaroon (Virginia Commonwealth Univ.)	
2-O-02	Surface Peening Technologies to Mitigate Stress Corrosion Cracking in Dry Cask Storage Stainless Steel Canister for Spent Nuclear Fuel	82
	Hwasung Yeom (POSTECH), John Lacy (EPRI), Kumar Sridharan (Univ. of Wisconsin)	
2-O-03	Experimental Evaluation of Performance for Gamma Emission Tomography to Inspect Partial-Defect Within Pressurized Water Reactor-Type Spent Nuclear Fuel	83
	Hyung-Joo Choi, Yoon Soo Chung, Yong Hyun Chung, Chul Hee Min (Yonsei Univ.), Hojik Kim, Sung-Woo Kwak, Heejun Chung (KINAC), Hee-Kyun Baek, Jung-ki Shin (NEOSIS Korea)	
2-O-04	AI Cathode Based Actinides Separation Over Lanthanides Via Molten Salt Electrolysis	84
	Weiqun Shi (Chinese Academy of Science)	
2-O-05	Long-Term Oxidation Behavior of Borated Stainless Steel (Neutron Absorber Based on Stainless Steel) in Spent Fuel Pool Environment	85
	Daehyeon Park, Yunju Lee, Junhyuk Ham, Ji Hyun Kim (UNIST)	
2-O-06	A Study on the Degradation Mechanism of Spent Fuels During Loss-of-Coolant-Accident Condition in a Spent Fuel Pool	86
	Sun-Ki Kim, Dong-Joo Kim, Jin-Hee Park (KAERI)	

2-O-07	Preliminary Thermal Analysis for the Design of an Experimental Device Aimed at Demonstrating the CANDU Spent Fuel Storage Silo	87
	TaeGang Lee, Jae Jun Jeong (Pusan Natl. Univ.), Taehyung Na (KHNP-CRI)	
2-O-08	Parameter Study of Radionuclide Release Rates for a Spent Fuel Transportation Cask Submerged in Deep Sea	88
	Guhyeon Jeong, Sanghoon Lee (Keimyung Univ.)	

2분과 사용후핵연료 처분전관리 [POSTER]

2-P-01	International Seismic Shake Table Tests for SNF Dry Storage Casks Update	91
	Hyeong Koo Kim, Joungyeul Lee, Seong Ki Lee (KEPCO NF), Ki Young Kim (KHNP-CRI), Elena Kalinina (Sandia Natl. Lab.)	
2-P-02	Preparation and Characterization of Metakaolin-based Sorbent Beads for Iodine Capture	92
	Jae Won Lee, Seok Min Hong, Byoungjin So, Chang Hwa Lee, Yung Zun Cho (KAERI)	
2-P-03	Adsorption Characteristics of Se-79 From Spent Nuclear Fuel on CaO Pellets ..	93
	Seok-Min Hong, Chang Hwa Lee, Namcheol Kim (KAERI)	
2-P-04	Reactive Distillation Process for the Strontium in Spent Nuclear Fuels Separated Via $MgCl_2$	94
	Eunsoo Lee, Sang Woon Kwon, Chang Hwa Lee (KAERI)	
2-P-05	Formation of K_2ZrCl_6 in LiCl-KCl Molten Salt	95
	Seungwoo Paek, Dalsung Yoon, Sang Kwon Lee, Chang Hwa Lee (KAERI)	
2-P-06	Evaluation of Alumina Scale Formed on Haynes 214 Alloy for Gas Corrosion Mitigation	96
	Su Jeong Heo, Sang-Kwon Lee, Jae Soo Ryu (KAERI)	
2-P-07	Comprehensive Studies on Characteristics of Spent Fuel From Small Modular Reactors Worldwide	97
	Ji Su Kang, Jae Hak Cheong (Kyung Hee Univ.)	
2-P-08	Applying the Slip Casting for Uranium Oxide Block Fabrication	98
	Yunmock Jung, Kweonho Kang, Sun Seok Hong, Chang Hwa Lee (KAERI)	
2-P-09	Design and Performance Test of Vertical Sintering Furnaces	99
	Yunmock Jung, Kweonho Kang, Sun Seok Hong, Seok Min Yoon, Chang Hwa Lee (KAERI)	
2-P-10	Self-Calibration of the Differential Scanning Calorimeter	100
	Kweonho Kang, Byeongjoo Yoon, Seokmin Yoon, Changhwa Lee (KAERI)	

2-P-11	Comparison of Base Metal & Weld Metal for Canister Chloride-Induced Stress Corrosion Cracking Using DCPD & EBSD Measurement	101
	Sang-Gyu Park, Ki-Hwan Kim, Sang-Soon Cho (KAERI)	
2-P-12	Hot Cell Examination for Measuring Rod Internal Pressure of Pressurized Dummy Fuel Rod	102
	Myungchul Park, Hyungkwon Lee, Dosik Kim (KAERI)	
2-P-13	Separation of Strontium Chloride From Uranium Oxide by Vacuum Distillation ..	103
	Sang Woon Kwon, Eunsoo Lee, Byungsuk Park, Joon Bo Shim, Chang Hwa Lee (KAERI)	
2-P-14	Temperature Evaluation of Spent Fuel Loaded in Dried Canning Container in the PIEF Pool	104
	Myungchul Park, Sukwoo Hong, Dosik Kim (KAERI)	
2-P-15	Thermodynamic Analysis of Residual Water Evaporation in Vacuum Drying of Spent Nuclear Fuel Canisters	105
	Ji Hwan Lim, Kyoung-Sik Bang, Seung-Hwan Yu (KAERI), Kyung-Wook Shin, Nam-Hee Lee (SAE-AN Eng. Corp.)	
2-P-16	Fabrication of Stabilized Pellets Using Fuel Powder Recovered From Damaged Fuel Rods by Vibratory Oxidative Decladding	106
	Jae Won Lee, Ju Ho Lee, ByoungJin So, Yung Zun Cho (KAERI)	
2-P-17	The Radiation Dose Assessment Based on the Neutron Shield Material in Spent Fuel Transfer Casks	107
	Donghee Lee (KHNP-CRI)	
2-P-18	Mass Spectrometry Calibration for Quantification of Residual Water Inside Spent Fuel Dry Storage Canisters	108
	Geon-Hui Lee, Suk-Won Lee, Kyung-Wook Shin, Gyung-Sun Chae, Jae-Seok Park (SAE-AN Eng. Corp.), Seung-Hwan Yu (KAERI)	
2-P-19	Vacuum Drying Technologies for Dry Storage of Spent Nuclear Fuel: A Comprehensive Review	109
	Ji Hwan Lim, Seung-Hwan Yu, Kyoung-Sik Bang (KAERI)	
2-P-20	Long-Term Integrity Verification of BSS (Borated Stainless Steel) Neutron Absorber	110
	Kiyong Kim, Donghee Lee, Beomgyu Kim (KHNP-CRI)	
2-P-21	Review of the NWTRB Report on the Disposal Evaluation of Spent Nuclear Fuel Stored in DOE's Dual-Purpose Canisters	111
	Donghee Lee, Yongdeog Kim, Kiyong Kim (KHNP-CRI)	
2-P-22	Preliminary Study of Cladding Integrity of Spent Fuel Using Probabilistic Safety Assessment	112
	Sangil Choi, Seunghwan Yu (KAERI)	

2-P-23	Structural Analysis of SNF Cladding Due to Corrosion-Induced Swelling of Pellet Under Residual Water Environment	113
	Sangil Choi, Sangsoon Cho (KAERI)	
2-P-24	Selection of Specimens for Evaluating Concrete Deterioration Properties in Spent Nuclear Fuel Dry Storage Facilities	114
	JaeHoon Lim, Si-wan Noh, Sang Soon Cho (KAERI)	
2-P-25	Spent Fuel Handling Guidelines at Nuclear Power Plants	115
	Sang-Gyoon Chang, Jun-Kyu Lee, Min-Gyu Kim (KEPCO E&C), Dae-Hyuk Kwon (KHNP)	
2-P-26	Establishment of Isotope Inventory DB to Develop Spent Fuel Burn-up Calculation Algorithm	116
	Seonghee Park, Younghwan Choi, YunSik Kim, Moonoh Kim (KONES Corp.)	
2-P-27	ISO 24389 Series and Its Utilization for Radioactive Waste Data Analysis Framework	117
	Kwangyoung Sohn, Changhwan Cho, Sungjong Kim, Kweonwoo Sohn (MIRAE-EN Co., Ltd.)	
2-P-28	Dissolution and Precipitation Behaviors of Barium Oxide in Molten Salts	118
	Seol Kim, Eun-Young Choi (UST), Chang Hwa Lee (KAERI)	
2-P-29	Layout Planning of Automated Pyroprocess Equipment in Hot Cells Based on the Relationship Between Equipment	119
	Jonghui Han, Yung-Zun Cho, Seungnam Yu (KAERI)	
2-P-30	Specific Heat Measurement With Differential Scanning Calorimeter	120
	Seokmin Yoon (UST), Kweonho Kang, Byeongjoo Yoon, Changhwa Lee (KAERI)	
2-P-31	A Study on the Long-Term Cooling Effect of Hydride Reorientation in Unirradiated Zircaloy-4 Cladding (12, 18 & 24 Months)	121
	Dae-Ho Kim, Hong-Ryoul Oh, Sung-Geun Kim, Dong-Hak Kook (KAERI)	
2-P-32	Development of CaO-based Adsorbents for $^{14}\text{CO}_2$ Capture	122
	Namcheol Kim (UST), Chang Hwa Lee, Seok-Min Hong (KAERI)	
2-P-33	Annular Metal Fuel Fabrication Technology Development Using Injection Casting Method	123
	Jungsu Ahn, Sang-Gyu Park, Seunguk Mun, Junhwan Kim (KAERI)	
2-P-34	Capture and Release of Oxygen in $\text{YBaCo}_4\text{O}_{7+x}$	124
	Byoungjin SO, Seok-Min Hong, Chang Hwa Lee (KAERI)	
2-P-35	A GoldSim Model for Mass Flow Analysis and Economic Evaluation of Back-end Fuel Cycle Incorporating Radionuclide Management Process	125
	Hong Jang, Hun Suk Im, Jungho Hur (KAERI)	

2-P-36	Reactor Design for Recovering the Metal Phosphates Precipitate by Segregation of the Salt Ingot With an Aid of a Dividing Plate Structure and Tapered Crucible ...	126
	Joon-Bo Shim, Sang-Woon Kwon, Chang Hwa Lee (KAERI)	
2-P-37	Applications of the Oxidation Treatment Method for the SF Volume Reduction in the SNF Dry Storage Facility	127
	Beomgyu Kim, Yongdeog Kim, Kyungho Roh, Kiyoun Kim (KHNP-CRI)	
2-P-38	Restoration of KOFA Spent Fuel Assembly	128
	Jae-Han Kim, In-Chan Kwon, Myungchul Park (KAERI)	
2-P-39	Development of Large Capacity CANDU Spent Nuclear Fuel Transport Cask (KTC-360)	129
	Woo-seok Choi, Yun-young Yang, Jongmin Lim, Kyung-sik Bang (KAERI), Seong-hoon Jeong, In-su Jeong (KONES Corp.)	
2-P-40	Comparison of Required Auxiliary Equipment for Concrete Overpack System ...	130
	Shin Dong Lee, Geon Woo Son, Chang Hee Han, Kwang Pyo Kim (Kyung Hee Univ.)	
2-P-41	Review on TLAA for Boron Depletion of Neutron Poison in Foreign Dry Storage Systems of Spent Nuclear Fuel	131
	JongYeol Baek, HyunSeok Song, Sang Hyeok Lee, Jae Hak Cheong (Kyung Hee Univ.)	
2-P-42	Estimation of Failure Parameters of RCT Simulation Model of Spent Nuclear Fuel Cladding Using Metamodel	132
	Seyeon Kim, Sanghoon Lee (Keimyung Univ.)	
2-P-43	Standardization Certification for SNF Burnup Calculation Methodology	133
	Ye Jeon Lim, Moonoh Kim (KONES Corp.)	
2-P-44	A Study on Characteristics of Ballooning Rupture of Spent Fuel Cladding Under Accident Condition	134
	Sun-Ki Kim, Dong-Joo Kim, Jin-Hee Park (KAERI)	
2-P-45	Analysis of Thermal Evaluations for Dry Storage Cask of High Burnup SF	135
	Taehyeon Kim, Donghee Lee, Sunghwan Chung, Yongdeog Kim (KHNP-CRI)	
2-P-46	Ventilation Control Analysis on SNF Dry Storage Facility Using Fire Modelling Tool	136
	Beomgyu Kim, Kyungho Roh, Kiyoun Kim (KHNP-CRI)	
2-P-47	Evaluation of Critical Flaw Sizes at Geometric Discontinuities in Spent Nuclear Fuel Storage Canister Using Finite Element Method	137
	Jongmin Lim, Sang Soon Cho (KAERI)	
2-P-48	Evaluation of the Radiological Impacts of Applying ICRP Publication 103 on Spent Fuel Transportation Casks	138
	Siwan Noh, Sang Soon Cho (KAERI)	

Withdrawal

2-P-49	A Study of Technical Standards of Overseas SF Dry Storage Facilities for the Design of SF Dry Storage Facilities in Korea	139
	JaeChan Park, TaeHoon Jeon, JungHo Song, JeeHyeon Kang, MinSu Ju (BRNC Inc.), JaeMin Lee (Orano Korea)	
2-P-50	Evaluation of Structural Integrity of HIC-510 L Container Based on the Technical Standards of Type a Transport Package	140
	Yongjoo Lee, Dongin Park, Jongbin Park, Sanghoon Lee (Keimyung Univ.)	
2-P-51	Preliminary Study for Transient CFD Simulation of Vacuum Drying	141
	Doyun Kim, Seunghwan Yu (KAERI)	
2-P-52	International Development Status of Wireless Monitoring Systems for Spent Fuel Dry Storage Facilities	142
	Donghee Lee, Taehyeon Kim, Taehyung Na (KHNP-CRI)	
2-P-53	A Study on Material Properties of Concrete in Dry Storage Facilities After Fire Test	143
	Seungpil Kim, Sang Soon Cho (KAERI)	
2-P-54	Review of Technical Requirements for Vacuum Drying Equipment for Nuclear Fuel Sealed Containers	144
	Suk Woo Hong, Myung Chul Park, In Chan Kwon, Sung Geun Kim, Do Sik Kim (KAERI)	
2-P-55	Development of Remote Visual Inspection System for Canister	145
	Taehyung Na, Yongdeog Kim, Donghee Lee (KHNP-CRI)	
2-P-56	First Interim Storage Examination of Bundles for CSF	146
	Taehyung Na, Yongdeog Kim, Donghee Lee (KHNP-CRI)	
2-P-57	Analysis of Test Result for Metallographic Examination of ISE	147
	Taehyung Na, Yongdeog Kim, Donghee Lee (KHNP-CRI)	
2-P-58	Development of the Evaluation Program Module for the Storage of SF	148
	Taehyung Na, Kiyong Kim, Beomgyu Kim (KHNP-CRI)	
2-P-59	Test Result of Dry Storage Inspection System for Canister (Silo Mock-up) ...	149
	Taehyung Na, Yongdeog Kim, Donghee Lee (KHNP-CRI)	
2-P-60	Requirements of Long-Term Dry Storage Experiment for CEX	150
	Taehyung Na, Yongdeog Kim, Donghee Lee (KHNP-CRI)	
2-P-61	Review of Dry Storage Application for the Development of SF Management Data System	151
	Taehyung Na, Kiyong Kim, Beomgyu Kim (KHNP-CRI)	

2-P-62	Review for Conceptual Scenario in Power Plant and Intermediate Storage of SF	152
	Taehyung Na, Beomgyu Kim (KHNP-CRI)	
2-P-63	Review of DOE's Work on the Disposition of SF in Dual-Purpose Canisters	153
	Taehyung Na, Donghee Lee, Kyungho Roh (KHNP-CRI)	
2-P-64	Review of DOE's Plans to Test Full-Size Spent Fuel Transportation Cask	154
	Taehyung Na, Donghee Lee, Kyungho Roh (KHNP-CRI)	

3분과 고준위폐기물처분 [ORAL]

3-O-01	Assessing Radiation-Induced Corrosion Damage on Copper Canisters in KAERI Reference Disposal System Using Numerical Simulation	157
	Nakkyu Chae, Jin-Seop Kim (KAERI), Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
3-O-02	Quantification of Risk Profile for Deep Geological Disposal Repository: A Comparison With TSPA	158
	Kibeom Son, Karyoung Choi, Gyunyoung Heo (Kyung Hee Univ.)	
3-O-03	Structural Investigation of Alkali Aluminoborosilicate Glass Containing MoO ₃ for Vitrification of Nuclear Waste	159
	Seon-jin Kim, Tae-min Yeo, Sung-hee Hyun, Jung-wook Cho (POSTECH)	
3-O-04	Radioactive Waste Management in Germany	160
	Toivo Wanne (BGE TEC)	
3-O-05	Effect of Different Copper Loadings for Efficient Iodine Capture on Metallic Copper Decorated Hexagonal Boron Nitride Under Dry and Humid Streams ..	161
	Tien-Shee Chee, Sujeong Lee, Ho Jin Ryu (KAIST)	
3-O-06	Impact of Organic Acids on the Dissolution of Iron-Uranium Simulated Fuel Debris	162
	Ryutaro Tonna, Takayuki Sasaki, Taishi Kobayashi (Kyoto Univ.), Yoshihiro Okamoto (JAEA)	
3-O-07	Adsorption Behavior of Iodate (IO ₃ ⁻) and Iodide (I ⁻) on Layered Double Hydroxides	163
	Sujeong Lee, Tien-Shee Chee, Ho Jin Ryu (KAIST), Louai Mahdi Maghrabi, Kyriaki Polychronopoulou (Khalifa Univ.)	
3-O-08	Assessment of Gas Generation and Transport in Geosphere: Implications for Deep Geological Repository Safety	164
	Jin-Seok Kim, Jang-Soon Kwon, Ji-Soo Kim, Soo-Hyeon Kim, Seung Yeop Lee, Sang-Ho Lee (KAERI)	

3분과 고준위폐기물처분 [POSTER]

3-P-01	IAEA's Roadmap for Implementing a Geological Disposal Programme: Facilitating Knowledge Transfer Between Advanced and Early-Stage Programmes	167
	Karina Lange, Stefan J. Mayer (IAEA), Haeryong Jung (KORAD)	
3-P-02	Current Status of the Radioactive Waste Management Program in Switzerland – Site Selection Process for a Deep Geological Repository	168
	Ingo Blechschmidt (NAGRA)	
3-P-03	Electrochemical Corrosion Behavior of Copper Fabricated by Additive Manufacturing for Deep Geological Disposal	169
	Gha-Young Kim, Jeong-Hyun Woo, Junhyuk Jang, Hyun Gil Kim, Young-Ho Lee, Yang-Il Jung, Seok Yoon (KAERI)	
3-P-04	Feasibility Analysis of Rod Consolidation for Spent Nuclear Fuel Disposal	170
	Samuel Park, Eunbi Cho, Jieon Kim, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
3-P-05	Alteration of Na Bentonite in YS03 Borehole	171
	Minsoo Lee, Jin-Seop Kim (KAERI), Mihye Kong (Chonnam Natl. Univ.)	
3-P-06	Preliminary Study on the Friction Stir Welds Characteristics of Oxygen-Free Copper for Spent Nuclear Fuel Disposal Canister	172
	Young-Ho Lee, Jeong-Mok Oh, June-Hyung Kim, Jin-Seop Kim (KAERI)	
3-P-07	Manufacturing of Engineering-Scale PWR Disposal Canister	173
	Kyungsub Han, Jongsul Park, Minsoo Jo, Yeonoh Lee (SeAH Besteel Corp.)	
3-P-08	Varying Disposal Depths in Deep Borehole Disposal of Spent Fuels	174
	Youn-Myoung Lee, Changsoo Lee, Dong-Keun Cho (KAERI)	
3-P-09	Considerations for the Bentonite Buffer Thickness in a Deep Geological Disposal of Spent Nuclear Fuels	175
	Jongyoul Lee, Heuijoo Choi, Dongkeun Cho (KAERI)	
3-P-10	Investigating Shear Behavior of Rock Joints in Granitic Rock Through Direct Shear Testing	176
	Seong Jun Ha, Taehyun Kim, Seungbeom Choi, Saeha Kwon, Jin-Seop Kim (KAERI)	
3-P-11	Damage-Coupled Hydro-Mechanical Modelling of Excavation Damaged Zone Development	177
	Ji-Won Kim, Chang-Ho Hong, Jin-Seop Kim (KAERI)	

3-P-12	Design of Bentonite Erosion and Piping Test (BEPT) in KAERI Underground Research Tunnel (KURT)	178
	Chang-Ho Hong, Seungbeom Choi, Jin-Seop Kim (KAERI), Eun-Soo Hong (HBC Inc.)	
3-P-13	Analysis and Design of Voloxidizer for Spent Fuel in Head-End	179
	Younghwan Kim, SunSeok Hong, Seokmin Hong, JuHo Lee, JaeWon Lee, ChangHwa Lee (KAERI)	
3-P-14	Assessment of Changes in Buffer Properties in an In-Situ Engineered Barrier Experiment of Korea	180
	Minseop Kim, Geon Young Kim, Jin-seop Kim, Seok Yoon (KAERI), Eun-soo Hong (HBC Inc.)	
3-P-15	Experimental Study on Mineralogical and Thermal-Hydraulic-Mechanical-Chemical Properties of Bentonite Purified Using Column Froth Flotation	181
	Minhyeong Lee, Seok Yoon, Gi-Jun Lee, Deuk-Hwan Lee, Seeun Chang (KAERI)	
3-P-16	Relationship Between Bentonil-WRK Bentonite Powder With Different Water Contents and Hyperspectral Imaging Data	182
	DeukHwan Lee, Seok Yoon (KAERI), Hwan-Hui Lim (KAIST)	
3-P-17	Temperature Variation of Near-Field Rock in Deep Geological Repositories for High-Level Radioactive Waste According to Thermal Conductivity of Buffer ...	183
	Gi-Jun Lee, Seok Yoon (KAERI), Min-Jun Kim (KIGAM), Gyu-Hyun Go (KIT)	
3-P-18	Site Characterization for Bentonite Erosion and Piping Test (BEPT) in KURT	184
	Seungbeom Choi, Chang-Ho Hong, Jaewook Ahn, Jin-Seop Kim (KAERI)	
3-P-19	A Preliminary Study on Using Two-Region and Two-Site Models for Describing Non-Equilibrium Transport Process in Apro	185
	Yong-Min Kim, Jung-Woo Kim (KAERI)	
3-P-20	Capability of the Numerical Analysis Model for the Near-Field Domain in APro	186
	Hong Jang, Heejae Ju, Jaewon Lee, Seungjin Seo, Gil-Eon Jeong, Soobin Kim, Jung-Woo Kim (KAERI)	
3-P-21	Defining Vegetation and Intake Data of ADioS for Evaluating Long-Term Evolution Scenarios Using Apro	187
	In-Young Kim, Minjeong Kim, Heejae Ju, Jung-Woo Kim (KAERI)	
3-P-22	Preliminary Study on the Effects of Temperature and Radiation on Monitoring Sensors for HLW Repository	188
	Sokpheanika Chea, Hyun-Joong Hwang, Changhee Park, Gye-Chun Cho (KAIST), Chang-Ho Hong (KAERI)	

3-P-23	Evaluation of the APro Reaction Module by Benchmark Simulation for Cation Exchange Reaction	189
	Jeonghwan Hwang, Jung-Woo Kim (KAERI), Weon Shik Han, Won Woo Yoon, Jiyong Lee (Yonsei Univ.)	
3-P-24	Procedure for Scenario-based Total System Performance Assessment by an Adaptive Process-based Total System Performance Assessment Framework (APro)	190
	Heejae Ju, In-Young Kim, Soobin Kim, Hong Jang, Jung-Woo Kim (KAERI)	
3-P-25	Designing a Benchmark Problem and Hydrological Modeling Based on Biosphere Scenarios	191
	Minjeong Kim, In-Yong Kim, Jung-Woo Kim (KAERI)	
3-P-26	Improvement and Verification for Near-Field Non-Isothermal Two-Phase Flow Model in Apro	192
	Jaewon Lee, Jung-Woo Kim (KAERI)	
3-P-27	Generation and Stability of Colloids From Compacted Bentonite Under the Synthetic Groundwater Condition	193
	Sang-Ho Lee, Jin-Seok Kim, Jisoo Kim, Soo Hyeon Kim, Seung Yeop Lee, Jang-Soon Kwon (KAERI)	
3-P-28	Radionuclide Transport Modeling Considering Adsorption in Excavation-Damaged Zone	194
	Jiyong Lee, Weon Shik Han, Won Woo Yoon (Yonsei Univ.), Jeonghwan Hwang, Jung-Woo Kim (KAERI)	
3-P-29	Thermodynamic Modeling of Calcium-Sodium Exchange Reactions in Bentonite-WRK Montmorillonite	195
	Seonggyu Choi, Hyun-Kyu Lee, Jae-Kwang Lee, Jeong-Yun Jang, Sang-Ho Lee (KAERI)	
3-P-30	Iodide Retention Mechanisms on Oxidized Cu Canister Under Expected Repository Conditions: Insights Into High-Level Radioactive Waste Disposal ...	196
	Ja-Young Goo, Seonggyu Choi, Seonyi Namgung, Seung Yeop Lee, Jang-Soon Kwon (KAERI), Yongheum Jo (Hanyang Univ.), Ho Young Jo (Korea Univ.)	
3-P-31	Geochemical Evaluation of Mafic Dykes in KURT Site: Implications for Long-Term Evolution of a Deep Geological Repository	197
	Nak Kyu Kim, Soolim Jung, Ji-Min Choi, Kyung-Woo Park (KAERI)	
3-P-32	Benchmarking of K-BANDI for Groundwater Release Into Agricultural Environments in a Constant Biosphere System	198
	Jonghyun Kim, Dongki Kim, Wontak Lee, Minhae Lee, Yisu Kim, Joowan Park (RADCORE)	

3-P-33	Development of Sorption and Diffusion Database Management System for the Safety Assessment of Radioactive Waste Disposal	199
	Jae-Kwang Lee, Seonggyu Choi, Hyun-Kyu Lee, Jang-Soon Kwon (KAERI)	
3-P-34	Development of Improvement Plan for Safety Assessment Complex Scenario Using Goldsim	200
	Hyun Jin Yu, Doo Yueb Kim, Jeong Hyo Oh, Yong Hee Han, Ye Ji Lee, Sang Ryeol Yoon, Jong Soon Song (Chosun Univ.)	
3-P-35	Using Goldsim to Create Concrete Deterioration Scenarios in Radioactive Waste Disposal	201
	Hyun Jin Yu, Min Ki Lee, Jun Yeong Park, Jeong In Choi, Eon Sang Jeon, Dong Gun Jang, Jong Soon Song (Chosun Univ.)	
3-P-36	Enhancing Disposal Efficiency: Criticality Analysis of Rod Consolidation for Spent Nuclear Fuel in Deep Geological Repository	202
	Eunbi Cho, Ji-eon Kim, Samuel Park, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
3-P-37	Processes of Long-Term Changes in the Surface Environment Due to Natural Climate Change, Uplift-Subsidence and Erosion-Deposition	203
	Nak-Youl Ko, Kyung-Woo Park (KAERI)	
3-P-38	Characteristics of Alteration in Jurassic Granitoids Using Boreholes Around KURT Site	204
	Soolim Jung, Ji-Min Choi, Nak Kyu Kim, Kyung-Woo Park (KAERI)	
3-P-39	Evaluation of Diffusion Characteristics of Radionuclides in Buffer Matrix Under Oxidation Conditions	205
	Hyun-Kyu Lee, Jae-Kwang Lee, Nak-Kyu Kim, Seonggyu Choi, Sang-Ho Lee, Jang-Soon Kwon (KAERI), Jaeun Kang, Hyojoo Kim, Wooyong Um (POSTECH)	
3-P-40	Investigating the Distribution of Radionuclides in Sedimentary and Granitic Aquifers Based on Age Tracers, Stable Isotopes, and Microbial Community Distribution	206
	YeoJin Ju, Kyung-Woo Park, Eunhye Kwon (KAERI), Dugin Kawon, Kang-Kun Lee (Seoul Natl. Univ.), Seong-Chun Jun, Jiwon Park (Geogreen 21)	
3-P-41	Demonstration Test of Emplacement and Retrieval Devices for Spent Nuclear Fuel in DBD	207
	Moonoh Kim, Hyun Woo Song, Duk Woon Jung (KONES Corp.), Dong Keun Cho (KAERI)	
3-P-42	Engineering-Scale Tracer Tests in a Natural Fractured Granite Rock	208
	Seonyi Namgung, Ki-Jong Jang, Yong-Kwon Koh, Ja-Young Goo, Seung-Hyo Son, Sang-Ho Lee, Jang-Soon Kwon (KAERI)	

3-P-43	Assessing the Influence of Chemical Composition of Solution and Temperature on the Zeolitization of Montmorillonite Using Natural Analogue Samples	209
	Wanhyoung Cho, Yoonah Bang, Ji-Hun Ryu (KAERI), Ho Young Jo (Korea Univ.)	
3-P-44	A Preliminary Study on the Applicability of Reactive Transport Model to Uranium Transport in Single-Well Push-Pull Test	210
	In-Hee Cho, Suh-Ho Lee, In-Woo Park, Jaeyeon Kim, Kang-Kun Lee (Seoul Natl. Univ.)	
3-P-45	Analysis of Fracture Aperture Evolution in Response to Stress Field Changes Using DEM-DFN Model	211
	Yong-Ki Lee, Chae-Soon Choi, Kyung-Woo Park (KAERI)	
3-P-46	Fault Rock Distribution in Drill Cores for Understanding Deep Brittle Structures Around the KAERI Underground Research Tunnel	212
	Ji-Min Choi, Soolim Jung, Nak Kyu Kim, Kyung-Woo Park (KAERI), Young-Seog Kim (Pukyong Natl. Univ.)	
3-P-47	Evaluating Corrosion Characteristics of Cu Between 25–75°C in Sulfide and Chloride Containing Solutions	213
	Seongkoo Hong, Samuel Park, Pilhyeon Ju, Taehoon Park, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.), Nakkyu Chae (KAERI)	
3-P-48	Solute Transport Visualization in Rough-Walled Rock Fractures Using Microscopic Imaging Technique	214
	Dahye Kim (KAERI), In Wook Yeo (Chonnam Natl. Univ.)	

4분과 중 · 저준위폐기물관리 [ORAL]

4-O-01	Dose Assessment for Residents in Off-Site for New Deployment of Radioactive Waste Incineration Facilities in Korea	217
	Hyeonjune Noh, Jaeyeong Park (UNIST)	
4-O-02	Assessment of Structural Stability and Leaching Characteristics of Phosphate-based Geopolymer Waste Form Containing Radioactive Spent Ion Exchange Resins	218
	Byoungkwan Kim, Hyun-min Ma, Seokyoung Oh, Wooyong Um (POSTECH), Solmoi Park (Pukyong Natl. Univ.), Brant Walkely (Univ. of Sheffield)	
4-O-03	A Study on Radioactive Residues From Tin Mining Industry in Indonesia	219
	Nurhadiansyah, Juyoul Kim (KINGS)	
4-O-04	Study for the Slope Stability of Engineering Barrier on Shallow Radioactive Waste Disposal Facility	220
	Chan Hong Kim, Hong-II Kwon, Beomin Kim (KORAD)	

4-O-05	The Worker's External Dose at Maintenance Work Due to Failure During Operation of the Spent Resin Mixed Waste Treatment Facility From Wolseong NPP Unit 1	221
	Ja Yeong Yoon, Hee Reyoung Kim (UNIST)	
4-O-06	PSR (Periodic Safety Review) Plan for Operating Gyeong-ju LILW Disposal Facilities	222
	Soojeong Ji, Jae-Yeol Cheong, Sungbok Lee (KORAD), Yongsuk Lee (FNC Technology Co., Ltd.)	
4-O-07	Non-Destructive X-ray Computed Tomography Characterization of Novel Wasteforms Targeting ¹⁴ C Solidification: Results of a Joint ROK-UK Project	223
	Richard I. Foster, Nakkyu Chae, Sungyeol Choi (Seoul Natl. Univ.), Tristan Lowe (Aletheia Imaging Solutions Ltd.), Kenny Watson, Steve Alderton (Waygate Technologies), Hyung-Ju Kim (KAERI)	
4-O-08	Mechanistic Investigation of Molybdenum Immobilized in Geopolymer Matrices	224
	Sajid Iqbal, Jong-Il Yun (KAIST)	
4-O-09	The Study on Optimal Solidification Conditions According to Composition Ratio of the Corrosion Products in the Sludge Waste From Nuclear Power Plant	225
	Ki Joon Kang, Hee Reyoung Kim (UNIST)	
4-O-10	Sampling Method Comparison for Homogeneity Assessment of Cementitious Radioactive Waste During Compressive Strength Measurement	226
	Hyeongjin Byeon, Ugyu Jeong, Jaeyeong Park (UNIST)	
4-O-11	Homogeneity Evaluation Test of Solid Waste Form Containing Spent Ion Exchange Resins Using Statistical Analysis	227
	Seokyoung Oh, Byoungkwan Kim, Hyun-min Ma, Wooyong Um (POSTECH)	
4-O-12	Effect of Si/Al Ratio in Metakaolin-based Geopolymer Formulation on Immobilization of Radioactive Selenium	228
	Hyun-min Ma, Byoungkwan Kim, Seokyoung Oh, Wooyong Um (POSTECH)	
4-O-13	Analysis of Solidified Paraffin Characteristics From Nuclear Power Plants and Consideration of Disposal Plans	229
	Young-Ku Choi, So Jung Shim, Seung Su Shin, Chang Heon Lee, Ji Sup Yoon (NDRI)	
4-O-14	Fabrication and Durability Evaluation of Cesium Pollucite Waste Form	230
	Ga Yeong Kim, Jae Hwan Yang (Chungnam Natl. Univ.), Ki rak Lee, Jung Hoon Choi, Jong Kwang Lee, Hwan Seo Park (KAERI)	

4-O-15	Ceramic and Cermet Waste Forms for Immobilization of Radioactive Sr Nuclide	231
	Byeonggwon Lee, Jung-Hoon Choi, Ki Rak Lee, Hyun Woo Kang, Hwan-Seo Park (KAERI)	

4분과 중 · 저준위폐기물관리 [POSTER]

4-P-01	Ultrasonic Decontamination Process to Treat Actual Uranium-Contaminated Gravel Under Neutral pH Condition	235
	Byung-Moon Jun, Hyejin Kim, Wooshin Park, Maengkyo Oh, Hongrae Jeon, San Chae, Hee-Chul Eun (KAERI)	
4-P-02	Direction for the Development of Thermal Treatment Technology for Intermediate- and Low-Level Radioactive Waste (ILRW)	236
	Seung-chul Park, Hee-jin Ahn (Green Radiation Co., Ltd.), Jun-ho Seo (Jeonbuk Natl. Univ.), Ji-hun Kim (KFE)	
4-P-03	Preliminary Study on the Uranium Precipitation Method for Wastewater Treatment From Soil Washing Processes Containing Uranium Isotope	237
	Maengkyo Oh, Keunyoung Lee, Byung-Moon Jun, Wooshin Park, JeongWook Moon, Jun Lee, Seung-Soo Kim, Hee-Chul Eun (KAERI)	
4-P-04	K-DWaT: An Inventory Verification Tool for Decommissioning Waste Using Microsoft Excel and Visual Basic Add-in	238
	Gun Young Yoon, Ji Hoon Han, Jae Hak Cheong (Kyung Hee Univ.)	
4-P-05	Comparative Study on Safety Standard History of IAEA Including GSG-18 for Clearance	239
	Hee Won Lee, Jae Hak Cheong (Kyung Hee Univ.)	
4-P-06	Development of Sample Handling and Transportation Processes for the Analysis of Radioactive Waste Characteristics in Hot Cell Facility	240
	Seonghee Park, Moonoh Kim (KONES Corp.)	
4-P-07	Radiochemical Decomposition of Oxalic Acid in Radioactive Liquid Waste	241
	Seung Joo Lim, Keunyoung Lee, Bum Kyung Seo, Mansoo Choi, Seonbyeong Kim, Wang Kyu Choi, Byeongseon Choi (KAERI), Wook Jae Yoo, Kyu Tae Park, So Yun Jeong, Hyun Young Shin (Orbitech Co., Ltd.)	
4-P-08	Understanding Other Laws for Clearance Level Radioactive Waste Focused on Chemically Decontaminated Uranium Contaminated Soil Waste	242
	YuJeong Choi, YunGun Jung, Chongyoung Cho, Seungjin Han, SangMo Hwang, Wonhyuk Jang (KAERI)	

4-P-09	A Study on the Radioactive Concrete Waste Reduction and Recycling Technology for the Decommissioning of Nuclear Power Plants	243
	Yong Guk Son, Jun-Yeol An, Min-Gyu Choi, Su-Young Choi (KRID)	
4-P-10	Evaluation of Applicability of Plasma Torch Melting for Perlite Heat Insulation Material	244
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI)	
4-P-11	Review on PE-HIC Generated From Contaminated Water Treatment Facility ..	245
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI)	
4-P-12	Case Study: A Bayesian Approach to Updating Scaling Factor for Radwaste	246
	Wook Sohn, Jieun Park, Kahee Jeong, Sooyeon Lim (KHNP-CRI)	
4-P-13	Study on the Cooling Hose of Vitrification Facility	247
	Seungmin Baek, Deukman Kim, Jung-Kwon Son (KHNP-CRI)	
4-P-14	Development of Copper Tellurite Glass Composition to Immobilize Radioactive Iodine	248
	Seong-Sik Shin, Young Hwan Hwang, Sunghoon Hong (KHNP-CRI), Hyun Woo Kang (KAERI)	
4-P-15	A Case Study on the Off-Gas Treatment System in Radioactive Waste Treatment Facilities	249
	Min-Gyu Choi, DoYeong Mun, Se Geun Lee, Yong-Won Ma, Yong Guk Son (KRID)	
4-P-16	A Case Study on Commercialization of Radioactive Waste Thermal Treatment Technology	250
	Min-Gyu Choi, DoYeong Mun, Se Geun Lee, Yong-Won Ma, Yong Guk Son (KRID)	
4-P-17	A Conceptual Design for Treatment of Large Concrete Drums in Kori Unit 1	251
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Seok-Ju Hwang, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI), Kwangsoo Park, Changgyu Kim, Chungsik Lim (Doosan Enerbility)	
4-P-18	KRID's Physical/Radiological Characterization Plan for Intermediate-Level Decommissioning Waste	252
	Junghyun Park, Seung-Ho Lee, Eunmi An, Seokhan Kim, Yeong Su Ha (KRID)	
4-P-19	A Conceptual Design for Pre-Treatment Equipment for Large Concrete Drums in Kori Unit 1	253
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Seok-Ju Hwang, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI), Kwangsoo Park, Chungsik Lim, Haewoong Kim (Doosan Enerbility)	

4-P-20	A Review on Treatment of Non-Fuel Materials in the SFP of PWR	254
	Kyungho Roh, Beomgyu Kim, Donghee Lee (KHNP-CRI)	
4-P-21	Experimental Study of the Adsorption Behavior of Iodine and Technetium on Dry Active Waste	255
	Yeojin Kim, Kiwon Kang, Yulim Lee, Jaeyeong Park (UNIST)	
4-P-22	Adsorption Behavior of Iodine and Technetium on Calcium-Silicate-Hydrated (C-S-H) Phases	256
	Yeojin Kim, Hyeongjin Byeon, Jaeyeong Park (UNIST)	
4-P-23	Status on Clearance of Decommissioning Waste at KRR-1 and -2	257
	Junhyuck Im, Doo-seong Hwang, Junhee Lee, Minyoung Kang, Geun-Ho Kim (KAERI)	
4-P-24	Safety Analysis on the High-Temperature Catalytic Oxidation Process for the Treatment of Spent Scintillation Liquid	258
	Jun-Seok Nam, Mikyoung Yang, Suil Bang, Young-Suk Oh (KHNP), Sung Jun Hong, Keon Hwa Lee (Hana NPE Co.), Woo Cheol Choi, Jung Ho Song, Tae Hoon Jeon (BRNC Inc.)	
4-P-25	Preliminary Study on an Uncertainty Evaluation of Chemical Pretreatment on Radioactive Waste	259
	Seongjin Maeng, Sang June Park, Jun-Gi Yeom (KHNP), Jieun Park (KHNP-CRI)	
4-P-26	A Preliminary Study on Deriving Implications for the Clearance of Decommissioning Waste in Korea Through International Case Studies	260
	Kinam Kwon, Seongjin Park, Jungho Song, Taehoon Jeon, Minsu Ju (BRNC Inc.)	
4-P-27	Bench-Scale Testing for the Development of a Commercial-Scale Paddle Type Heating-Grinding Integrated Reactor	261
	Hyun-Young Shin, Ba-Ro Lee, Kyu-Tae Park, Chan-Hee Park, Sung-Ryul Kim, Jung-Min Oh, Ji-Ung Kim, Hye-Jin Kim, So-Yun Jeong, Sang-Mook Kang, Oh-il Kwon, Wook Jae Yoo (Orbitech Co., Ltd.), Keun-Young Lee, Jaewoong Hwang (KAERI)	
4-P-28	Thermodynamic Equilibrium Calculations of Treatment of Radioactive Wastewater Containing Uranium	262
	Hee-Chul Eun, Maengkyo Oh, Byung-Moon Jun, Woo-Shin Park, Tack-Jin Kim, Sung-Bin Park, Sunju Oh (KAERI)	
4-P-29	A Study on the Management Plan of Radioactive Waste Management System for Continued Operation of Nuclear Power Plant	263
	JunKi Baik, HyeJin Jung, HeeMoo Heo (KHNP-CRI)	

4-P-30	Development of Prototype of Handling Device for Large Concrete Drums in Kori Unit 1	264
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI), Kwangsoo Park, Changgyu Kim, Chungsik Lim, Haewoong Kim (Doosan Enerbility)	
4-P-31	Development of on Evaluation Platform to Advance Radioactive Waste Solidification Analysis Technology	265
	Hee-Jin Ahn, Seung-Chul Park (Green Radiation Co., Ltd.), Dae-Seok Hong (KAERI)	
4-P-32	A Study on the Repackaging of Solidified Concentrated Liquid Radioactive Waste Into PC-HICs	266
	Changgyu Kim, Kwangsoo Park, Chungsik Lim, Haewoong Kim, Taeseob Lim, Donghun Pak (Doosan Enerbility), Jungkwon Son, Younghwan Hwang, Sunghoon Hong (KHNP-CRI)	
4-P-33	Effect of Salt in Radioactive Wastewater on the Stability of Cement Waste Form	267
	Jun Young Jung, Somi Lee, Naon Chang, Duckha Kim, Dong-Su Kim, Tack-Jin Kim (KAERI)	
4-P-34	Introduction of the Developed Paraffin Solidified Concentrate Waste Treatment Method and System	268
	Hyun Woo Song, Moonoh Kim (KONES Corp.), Hwan Seo Park (KAERI)	
4-P-35	Experimental Study of Dissolution of Glass Fiber Waste in the Domestic Repository and Its Estimated Colloidal Impact	269
	Hyun Woo Song, Moonoh Kim (KONES Corp.), Sang June Park, Sungjun Kim, Jun-gi Yeom (KHNP)	
4-P-36	A Study on Material Integrity Through Biodegradability of PE-HIC	270
	Jisoo Yoon, Young Hwan Hwang, Jung-Kwon Son (KHNP-CRI)	
4-P-37	A Study on Overseas Disposal Cases of PE-HIC	271
	Jisoo Yoon, Young Hwan Hwang, Jung-Kwon Son (KHNP-CRI)	
4-P-38	A Study on Technical Standards for Packaging Spent Filters by UK NIREX	272
	Jisoo Yoon, Deuk-Man Kim, Jung-Kwon Son (KHNP-CRI)	
4-P-39	A Study on Effect of Hf Addition in DG-2 Glass	273
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Seok-Ju Hwang, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-40	Solidification of Soil Containing Boric Acid Using Portland Cement	274
	Somi Lee, Naon Chang, Duckha Kim, Jun Young Jung, Dong-su Kim (KAERI)	
4-P-41	Structural Analysis of Copper Tellurite Glass for Radioactive Iodine Immobilization	275
	Seong-Sik Shin, Young Hwan Hwang, Sunghoon Hong (KHNP-CRI), Hyun Woo Kang (KAERI)	

4-P-42	A Review of Geopolymer Properties According to Composition and Reaction Conditions	276
	Seong-Sik Shin, Young Hwan Hwang, Sunghoon Hong, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-43	A Study on Solidification of Radioactive Waste From Nuclear Power Plant	277
	Seong-Sik Shin, Young Hwan Hwang, Sunghoon Hong, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-44	A Review of Geopolymer Waste Form for Radioactive Waste Immobilization ..	278
	Seong-Sik Shin, Young Hwan Hwang, Sunghoon Hong, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-45	A Study on Re-Solidification of Inner Waste in Large Concrete Drums	279
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Ji-Soo Yoon, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI), Changgyu Kim, Donghun Park, Taeseob Lim (Doosan Enerbility)	
4-P-46	Development of Over-Packaging Container for Polyethylene High Integrity Container (PE-HIC) in Korea	280
	Ji-Hoon Lee, Young-Hwan Hwang, Seong-Hoon Hong, Ji-Soo Yoon, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-47	Structural Evaluation of Improved Disposal Canister Emplacement and Retrieval Device Under Lifting Condition	281
	Young-Oh Lee, Moonoh Kim, Tae-Su Kim (KONES Corp.), Dong-Keun Cho (KAERI)	
4-P-48	Review on the Design of Cold Crucible Induction Melter for Radioactive Wastes	282
	Deuk-Man Kim, Seung-Min Baek, Jung-Kwon Son (KHNP-CRI)	
4-P-49	Management of Radioactive Solid Waste From the HANARO in 2023	283
	Kyungsu Park, Youngsan Choi, Wootae Lee, Minsu Kim (KAERI)	
4-P-50	Review on the Validation of Activated Material Modeling Analysis	284
	Seok-Ju Hwang, Younghwan Hwang, Jisoo Yoon, Jung-Kwon Son (KHNP-CRI)	
4-P-51	Review on the Indirect Evaluation Methodology for the Radionuclide Inventory of the Dry Active Waste (DAW) Drums Generated in the New NPPs	285
	Hang-Rae Cho, Ji-Soo Yoon, Young-Hwan Hwang, Jung-Kwon Son (KHNP-CRI), Seong-Jin Maeng (KHNP)	
4-P-52	Review of Latest Radioactive Waste Monitoring Technology	286
	Seok-Ju Hwang, Younghwan Hwang, Jisoo Yoon, Jung-Kwon Son (KHNP-CRI)	
4-P-53	Status Update of the Progress of the Hanford WTP Vitrification in US DOE Hanford Site	287
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI)	

4-P-54	Status Update of LLW Disposal Facilities and PE-HIC Disposal for Radioactive Wastes in the United States	288
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI)	
4-P-55	Recent Status for Contaminated Water Management Generated From Fukushima Daiichi NPP	289
	Sunghoon Hong, Young Hwan Hwang, Seong-Sik Shin, Jeongsu Jeong, Jung-Kwon Son, Cheon-woo Kim (KHNP-CRI)	
4-P-56	Japanese Practice of Assessing Radioactive Inventory of Radwaste Focusing on Mean Activity Concentration Method	290
	Wook Sohn, Jieun Park, Kahee Jeong, Sooyeon Lim (KHNP-CRI)	
4-P-57	A Case Study on Activation Calculation of Non-Fuel Materials in the SFP of PWR	291
	Kyungho Roh, Beomgyu Kim, Donghee Lee, Hongki Seo (KHNP-CRI)	
4-P-58	A Case Study on Nuclide Analysis of Non-Fuel Materials in the SFP of PWR ...	292
	Kyungho Roh, Beomgyu Kim, Donghee Lee, Hongki Seo (KHNP-CRI)	
4-P-59	Preliminary Study of Radiation Shielding Analysis for ILW Hot Cell Workers ...	293
	Seung Uk Yoo, Moonoh Kim (KONES Corp.)	
4-P-60	Development of ODCM and Its Application to Operational Dose Assessment of Gyeongju LILW Repository	294
	Jeong Hyoun Yoon, Hye Yeong Min, Ki Dae Han, Eun Sang Park, Joo Ho Oh (KORAD)	
4-P-61	Calculation of Radioactive Sensitivity According to Irradiated Stainless Steel-304 Composition Tolerance	295
	Seok Geun Cho, Hyuk Han, Chang-Je Park (Sejong Univ.)	
4-P-62	A Review on International Radioactive Waste Management and Disposal Facilities	296
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Ji-Soo Yoon, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-63	A Review on Radioactive Waste Management of Japan	297
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Seok-Ju Hwang, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
4-P-64	A Preliminary Study on Improving Classification and Management Methods for Radioisotope Waste in Korea	298
	Seongjin Park, Jungho Song, Ki Nam Kwon, Taehoon Jeon, Minsu Ju (BRNC Inc.)	

4-P-65	Directions to Improve Public Acceptance in a Host City of High Level Waste Disposal Facility–Final Repository	299
	Ki-jung Kwon, Hyun Soo Kim, Sangki Lee, Jae Youn Lee, Jaemin Lee, Changhoon Song, Min Hye Hwang, Seok Won Choi, Chi-Hwan Go (KORAD)	
4-P-66	Introduction to Development and Commercial Operation of Vitrification Plant China (VPC) for HLW Treatment	300
	Cheon-Woo Kim, Young Hwan Hwang, Sunghoon Hong (KHNP-CRI)	
4-P-67	Preliminary Disposal Cost Assessment According to Spent Activated Carbon Processing Method	301
	Seung Uk Yoo, Moonoh Kim (KONES Corp.)	
4-P-68	Inspection of Radiocative Waste Through 4 th Industrial Revolution Technology ..	302
	Junsung Son, Jeong-Woo Yun, Ga-Young Cha, Ji-Hyeon Choe, Hyewon Shin (KORAD)	
4-P-69	Preliminary Consideration of Safety Assessment for a Landfill Disposal Facility Using RESRAD-ONSITE Code	303
	Younghoon Hwang, Jae Hak Cheong (Kyung Hee Univ.)	
4-P-70	A Systematic Approach to Improving Regulations to Manage Mixed Waste Efficiently	304
	Moonoh Kim, Hyun Woo Song (KONES Corp.), Sang June Park, Sungjun Kim, Jun-gi Yeom (KHNP)	
4-P-71	Proposal of Concept for On-Site Disposal of Decommissioning Waste From NPPs	305
	In Gyu Chang, Jae Hak Cheong (Kyung Hee Univ.)	
4-P-72	Improving Safety and Efficiency of Radioactive Waste Disposal Through Coating Application	306
	Ji-Hoon Kang, Dong-Seok Lim, Woo Young Jung, Hyun Chul Lee, Seung Heon Baek (FNC Technology Co., Ltd.)	
4-P-73	Review of Overseas Radwaste Disposal Site WAC	307
	Seok-Ju Hwang, Younghwan Hwang, Jisoo Yoon, Jung-Kwon Son (KHNP-CRI)	
4-P-74	Case Study on the Final Disposal of Radioactive Concrete Waste at KAERI ...	308
	Dong-Ju Lee, Ill-Sik Kang, Duck Ha Kim, Hyo Jong Kim, Dae Seong Nam, Jong Hwa Park, Hae Min Park, Yunjeong Hong, Jong Soo Nam, Won Hyuk Jang (KAERI)	
4-P-75	A Conceptual Design for Disposal of PE-HIC in Disposal Facility	309
	Young Hwan Hwang, Sunghoon Hong, Ji-Soo Yoon, Seong-Sik Shin, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	

4-P-76	Preliminary Analysis of Characterization Terms to Derive a Disposal Concept (Draft) for Long-Term Storage RI Sealed Source	310
	Hoseog Dho, Seho Choi, Yuhwa Son, Jaechul Ha (KORAD)	
4-P-77	Proposal of Design Requirements for the Development of Intermediate Level Waste Containers	311
	Narae Lee, Moonoh Kim (KONES Corp.)	
4-P-78	A Review of the Scenario Selection Method of the Radioactive Waste Disposal Facility	312
	Sang June Park, Seongjin Maeng, Jun-Gi Yeom (KHNP), Hyun Woo Song, Moonoh Kim (KONES Corp.)	
4-P-79	A Study on the Statistical Similarity of Waste Streams in Radioactive Waste Characteristics	313
	Yu-Hwa Son, Jae-Chul Ha, Myung-Goo Kang (KORAD)	
4-P-80	Manufacturing and Performance Test Method of Concrete Disposal Container	314
	Seong Ha Yun, Jung Woo Jo, Woo Yeol Yoon, Chung Hyeok Kim, Joo Ho Oh (KORAD)	
4-P-81	Introduction of Piggling System for Ensuring the Safety of the Groundwater Drainage System at the Gyeongju Underground Silos Disposal Facility	315
	Changyeol Choi, Jaejun Kim (KORAD)	

5분과 제염해체 [ORAL]

5-O-01	Spatial Dose Evaluation of Decommissioning Kori NPP Unit 1 Bioshield	319
	Jaeho Lee, Hyeonmin Lee, Seungbin Yoon, Woonyun Choi, Hee Reyoung Kim (UNIST)	
5-O-02	Fabrication and Characterization of Thin Plastic Scintillator for Beta/Gamma Detection	320
	Su Jung Min, Si On Kim, Sang Bum Hong, Chang Hyun Roh, Bum Kyoung Seo (KAERI)	
5-O-03	Comparison of Calculated and Measured Radioactivity Values of Kori Unit 1 Baffle Former Bolt	321
	Young-Jae Maeng, Hyun-Chul Lee, Myung-Ho Lee, Choon-Sung Yoo (KRIST), Seung-Jin Oh (KHNP-CRI), Sung-Sik Hwang (KAERI)	
5-O-04	Research Status of Laser Technology Development for Cutting Activated Structures in Nuclear Power Plants	322
	Moo-Keun Song, Dong-Hyun Kim, June-Hyun Kim, Do-Yeob Kim, Sun-Il Kim (KRID), Su-Jin Lee (KIMM), Jong-Do Kim (Korea Maritime & Ocean Univ.)	

5-O-05	Developing a Deep Learning Model for Predicting Internal Decontamination Levels in Radioactively Contaminated Tanks	323
	Sia Hwang, Hee Reyoung Kim (UNIST)	
5-O-06	Analysis of the Binding Characteristics Between Radionuclides and Soil Using Sequential Extraction Method	324
	In-Ho Yoon, Sunyoung Park, Ilgook Kim (KAERI)	
5-O-07	Stabilization of Combustible Waste Carbonization Residues Through Zeolite Precipitation and Sintering Technology	325
	Jaewoong Hwang, Jaseung Koo (Chungnam Natl. Univ.), Keunyoung Lee (KAERI)	
5-O-08	An Analysis on Guidance for Management of Hazardous Materials During Decommissioning Nuclear Power Plants	326
	Jihwan Yu, Hyunjun Jo, Minchul Kim, Hyu-chang Choi, Sukwon Jung (KHNP-CRI)	

5분과 제염해체 [POSTER]

5-P-01	Development of an Intelligent IoT Sensor System for Real-Time Radiation Measurement	329
	Myung-sub Roh, Sang-jin Lee, Sang-bok Kim, Ki-kyung Noh (Daekeong Eng. Co., Ltd.)	
5-P-02	Evaluation of Internal Exposure of Wolsong Unit 1 Reactor Decommissioning ..	330
	Min Chul Kim, Gang Woo Ryu, Young Il Na (KHNP-CRI)	
5-P-03	Comparison of Dose Evaluation Methodology for PHWR Decommissioning ..	331
	Min Chul Kim, Gang Woo Ryu, Young Il Na (KHNP-CRI)	
5-P-04	Selection Process of Potential Radionuclides for the Decommissioning of Wolsong Unit 1	332
	Kwangho Jo, Kangwoo Ryu, Youngil Na (KHNP-CRI)	
5-P-05	Hydrogen Isotope Separation Behavior of Porous Media and Structural Stability Under Beta Irradiation	333
	Hye Jin Cho, Euna Jeong, Chan Woo Park (KAERI)	
5-P-06	Investigation on Generation Characteristics of Radioactive Air Pollutant and Air Purification Technology During Nuclear Power Plant Decommissioning	334
	Kibong Nam, Chanyoung Lee, Moonjo Han, Seongwoo Lee (WOOJIN Co., Ltd.)	
5-P-07	An Evaluation Method for Radiological Characteristics to Assess Performance of System Decontamination	335
	Seonghun Kim, Younjung Choi, Hyerim Kim, Jihyun Yu, Jangsoo Suh (Sae-An Enertech Corp.)	

5-P-08	Analysis of the Impact of Nb-94 on the Selection of Radionuclides of Concern for Preparing Wolsong Unit 1 Decommissioning	336
	Dong-Gyun Lee, SeokYoung Ahn (Pusan Natl. Univ.), Soon-Hwan Son (AGENCORE Co., Ltd.)	
5-P-09	A Study on the Decay Correction Factor for Comparison With Overseas Cases in the Dose Estimation of Decommissioning Workers	337
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
5-P-10	Review of the Radionuclides of Interest by Media Considered at the Time of License Termination for the Decommissioning Nuclear Power Plants	338
	Young-II Na, Hyung-Woo Seo, Gang-Woo Ryu (KHNP-CRI)	
5-P-11	Fundamental Study on Cutting of Zirconium Alloys With a High-Power Fiber Laser for Decommissioning of Heavy Water Reactors	339
	Jae Sung Shin (KAERI)	
5-P-12	A Study on the Aerosol and Particle Collection Method and Device of Underwater Operations During Laser Cutting of Activated Structures Around the Reactor	340
	HaeWoong Kim, Kwangsoo Park, Changgyu Kim, Hoseok Nam, Chungsik Lim, DongHun Pak, TaeSeob Lim (Doosan Enerbility)	
5-P-13	Study of the Dismantling Sequence of Wolseong Unit 1 Reactor Cooling System From a Radiation Safety and Economic Perspective	341
	Taeseob Lim, Haewoong Kim, Kwangsoo Park, Chungsik Lim, Changgyu Kim, HoSeok Nam, DongHun Pak (Doosan Enerbility)	
5-P-14	Plasma Processing Technology Development and Operation Status	342
	Jeongsu Jeong, Sunghoon Hong (KHNP-CRI)	
5-P-15	A Status of the Global Demonstration Facilities of Advanced Decommissioning Equipments	343
	JiHoon Han, Do young Moon, Sun il Kim (KRID)	
5-P-16	A Case Study on Considerations When Assessing Decontamination Technologies for Decommissioning	344
	Do-Yeob Kim, Dong-Hyun Kim, June-Hyun Kim, Moo-Geun Song (KRID)	
5-P-17	Case Study of Aerosol Production and Radionuclide Contamination in Decommissioning of NPP	345
	Donghun Pak, Haewoong Kim, Kwangsoo Park, Chungsik Lim, Changgyu Kim, Hoseok Nam, Taeseob Lim (Doosan Enerbility)	

5-P-18	Isotope Selective Electrical Transmission of Hydrogen (H/D) in Liquid Electrolyte Through PTFE Reinforced PFSA Polymer–Graphene Composite Membrane	346
	Hyojoo Kim, Wooyong Um (POSTECH)	
5-P-19	Investigating Metal Oxide Solubility in Choline Chloride–based Deep Eutectic Solvents for Potential Selective Spent Catalyst Recovery	347
	Kwangseo Kim, Stuart Aberdeen, Shuang Liu, Woohyun Lim, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
5-P-20	Electrochemical Dissolution of Metal Oxides in Deep Eutectic Solvents for Decontamination Applications	348
	Woohyun Lim, Shuang Liu, Stuart Aberdeen, Kwangseo Kim, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
5-P-21	Fabrication of a Nanofibrous Pt/Ir–RuO ₂ Electrode for Treatment of Organic Decontamination Waste	349
	Chanhee Jeong, Tae Hyeon Kim, Kune–Woo Lee, Taek Seung Lee (Chungnam Natl. Univ.)	
5-P-22	Prussian Blue Decoration on the Polyethylenimine Grafted Polyacrylonitrile Nanofiber for the Enhanced Removal of Radioactive Cesium From Water	350
	Ga–Eun Lee, Kyung–Jin Lee (Chungnam Natl. Univ.), Ye–Won Jeong, Hee–Man Yang (KAERI)	
5-P-23	The Decontamination Process of Radioactive Metal Waste by Using Inorganic Based Chemical Agent	351
	Sung Hwan Cho, Mansoo Choi, Heung June Chung, Wang–Kyu Choi, Seung Joo Lim (KAERI)	
5-P-24	Removal of Radioactive Cesium From Surfaces Using Partially Hydrolyzed Poly (Vinyl Acetate)–Borax Complex Containing Cs Adsorbent at Low Temperatures	352
	Ye–Won Jeong, Hyung–Ju Kim, Hee–Man Yang (UST), Ga–Eun Lee (Chungnam Natl. Univ.), Sung–Wook Kim (KAERI)	
5-P-25	The Removal of Metals [Cd(II), Ni(II), Sb(V)] by Using Metal Oxide Biochar Composite	353
	Jeong–Yun Jang, Sang–Ho Lee (KAERI), Dong–Wan Cho, Chul–Min Chon, Gil–Jae Yim (KIGAM)	
5-P-26	Effect of Catalyst Deposition Methods on the Long–Term Performance of PEM Electrolyzers for Hydrogen Isotope Separation	354
	Euna Jeong, Kyung Jin Lee (Chungnam Natl. Univ.), Chan–Woo Park (KAERI)	
5-P-27	Copper Hexacyanoferrate as an Efficient Capacitive Deionization Electrode for the Removal of Cobalt Ions From Wastewater	355
	Sang–Hun Lee, Jeongmook Lee, Jong–Yun Kim, Sang Ho Lim (KAERI)	

5-P-28	The Technology Evaluation Factors and Special Considerations for the Environmental Remediation of Radioactively Contaminated Sites	356
	KwanSeong Jeong, Geun-Ho Kim, Ilgook Kim, ChanWoo Park, Youngho Sihm, In-Ho Yoon (KAERI)	
5-P-29	Comparison and Analysis of DCGLs for a Single NPP Site and Multiple NPPs Site Using RESRAD-ONSITE	357
	Sang Hyeok Lee, Jae Hak Cheong (Kyung Hee Univ.)	
5-P-30	Analysis of Media Subject to DCGL Derivation for Overseas Decommissioning Nuclear Power Plants	358
	Gang Woo Ryu, Hyung Woo Seo, Young Il Na, Chan Geun Park (KHNP-CRI)	
5-P-31	A Study on Developing Elements of Radiation Scanning Robotics Aligning With the MARSSIM Revision	359
	Yong-Won Ma, Se Geun Lee, Su Young Choi, Hyeok Jae Kim (KRID)	
5-P-32	A Preliminary Study on Considerations of Measurement Quality Objectives for Final Status Survey Based on MARSSIM Revision	360
	Se Geun Lee, Hyeok Jae Kim, Yun Hui Lee (KRID)	
5-P-33	Predicting the Distribution Coefficients of Strontium and Cobalt Using Hybrid Method With Machine Learning and Experimental Data	361
	In-Ho Yoon, Ilgook Kim (KAERI), Seongyeon Na, Seok Min Hong (UNIST), Kyung Hwa Cho (Korea Univ.)	
5-P-34	Evaluation of DCGL for Clearance of Spent Activated Carbon Using RESRAD-Onsite Code	362
	Seongmin Jo, Taewoong Lee, Kihyun Cho, Kichul Kim, Seunghyun Lee, Junghyun Lee (KEPCO KPS)	
5-P-35	Effects of Dose Evaluation by Exposure Pathways According to the Site Reuse Scenario Used in Decommissioning Nuclear Power Plants	363
	Hyung-Woo Seo, Young-II Na, Chan-Geun Park, Gang-Woo Ryu (KHNP-CRI)	
5-P-36	Analysis of Site Conceptual Model of Basement for Overseas Decommissioning Nuclear Power Plants	364
	Gang Woo Ryu, Hyung Woo Seo, Young Il Na, Chan Geun Park (KHNP-CRI)	
5-P-37	Preliminary Analysis of the Characteristics of Waste Generation From Specific Decommissioning Activities	365
	Su Yeon Lee, Jae Hak Cheong (Kyung Hee Univ.)	
5-P-38	A Review on the Management of Mixed Waste for Decommissioning of Nuclear Power Plants	366
	Suk-Won Jung, Hyung-Woo Seo (KHNP-CRI)	

5-P-39	An Overview of Asbestos Abatement for Plant Decommissioning	367
	Jihwan Yu, Hyunjun Jo, Minchul Kim, Hyu-chang Choi, Sukwon Jung (KHNP-CRI)	
5-P-40	Current Status of Detailed Design of Radioactive Waste Treatment Facilities in Preparation for Decommissioning of Nuclear Power Plants	368
	Suk Bon Yoon, Hyun Min Kim, Sukwon Jung, Gangwoo Ryu (KHNP-CRI)	
5-P-41	Development of the Metal Incorporated Mesoporous Silica Adsorbents for Hydrogen Gas Separation	369
	Hyung-Ju Kim, Daeuk Kang, Chanwoo Park (KAERI)	
5-P-42	Analysis of Recycling Methods as a Solidification Agent for Radioactive Concrete Waste Generated During Nuclear Power Plant Decommissioning ..	370
	Jun-Yeol An, Yong-Guk Son, Min-Gyu Choi, Su-Young Choi (KRID)	
5-P-43	Characterization of the Mixed-Waste Generated for Decommissioning of Kori Unit 1	371
	HyunMin Kim, SukWon Jung, GangWoo Ryu, Suk Bon Yoon (KHNP-CRI)	
5-P-44	Development of Radiation Shielding Equipment for Safe Handling of Large Concrete Drums Containing Radioactive Waste	372
	Chungsik Lim, Kwangsoo Park, Changgyu Kim, Haewoong Kim, Hoseok Nam, Taeseob Lim, Donghun Pak (Doosan Enerbility), Jungkwon Son, Younghwan Hwang, Sunghoon Hong (KHNP-CRI)	
5-P-45	A Preliminary Study on GUI Optimization for Decommissioning and Waste Management Software Based on 4 th Industry Revolution	373
	Se Geun Lee, Yong-Won Ma, Hyeok Jae Kim (KRID)	
5-P-46	Radioactive Waste Management of Wolsong Unit 1 Reactor Decommissioning ..	374
	Min Chul Kim, Gang Woo Ryu, Young Il Na (KHNP-CRI)	
5-P-47	Characteristics of Liquid Radioactive Waste in the Case of Domestic Nuclear Facility Decommissioning	375
	Ki-Baek Shin, Dooseong Hwang, Dae-Seok Hong (KAERI), Gyuseong Cho (KAIST)	
5-P-48	Development of a High-Speed Radioactive Liquid Waste Treatment System Using Nanocatalysts	376
	So-Yun Jeong, Hyun-Young Shin, Wook Jae Yoo, Chan-Hee Park, Sung-Ryul Kim, Jung-Min Oh, Ba-Ro Lee, Ji-Ung Kim, Hye-Jin Kim, Kyu-Tae Park (Orbitech Co., Ltd.), Seung-Joo Lim (KAERI)	
5-P-49	Prospects of Temporary Storage and Final Disposal for Intermediate-Level Radioactive Waste From Decommissioned Nuclear Power Plants	377
	Cheon-Woo Kim, Sunghoon Hong, Young Hwan Hwang (KHNP-CRI)	

5-P-50	Treatment and Volume Reduction Plan for Kori Unit 1 Decommissioning Radioactive Waste By Using Induction Melter and Plasma Torch Melter	378
	Cheon-Woo Kim, Sunghoon Hong, Young Hwan Hwang (KHNP-CRI)	
5-P-51	Development of Remote Sampling Device for Radioactive Waste Filter of Nuclear Power Plant	379
	Ji-Hoon Lee, Kyung-Rok Park, Ji-Soo Yoon, Jung-Kwon Son (KHNP-CRI), Moon-Oh Kim, Na-Rae Lee (KONES Corp.)	
5-P-52	Assessment of the NORM (TiO ₂) Waste Disposal Option in Korea Industry ...	380
	Sang Bum Hong, Su Jung Min, Bum Kyoung Seo (KAERI), Zu Hee Woo (KINS)	
5-P-53	A Review on Recycling and Reuse of Materials From Decommissioning of Nuclear Facilities	381
	Jihwan Yu, Hyunjun Jo, Minchul Kim, Hyu-chang Choi, Sukwon Jung (KHNP-CRI)	
5-P-54	A Review on Management of the U.S. Nuclear Plant Buried and Underground Piping	382
	Young-II Na, Hyung-Woo Seo, Chan-Geun Park, Gang-Woo Ryu (KHNP-CRI)	
5-P-55	Considerations for Root Causes and Contingency Management of Abnormal Situations During Decommissioning of Nuclear Facilities	383
	KwanSeong Jeong (KAERI), Hyerim Kim, Jangsoo Suh, Jihyun Yu, Younjung Choi (Sae-An Enertech Corp.)	
5-P-56	A Method for Calculating the Preliminary Removal Cost for the Decommissioning of Domestic Nuclear Power Plant	384
	Chan-Geun Park, Hyung-Woo Seo (KHNP-CRI)	
5-P-57	A Review of Influencing Factors for Selecting the Decommissioning Strategy	385
	Hyung-Woo Seo, Young-II Na, Chan-Geun Park, Gang-Woo Ryu (KHNP-CRI)	
5-P-58	A Review of BIM Technology Application for Decommissioning NPP	386
	Young-II Na, Hyung-Woo Seo, Chan-Geun Park, Gang-Woo Ryu (KHNP-CRI)	
5-P-59	Analysis of ALARA Evaluation Methodology for Overseas Decommissioning Nuclear Power Plant	387
	Gang Woo Ryu, Hyung Woo Seo, Young II Na, Chan Geun Park (KHNP-CRI)	
5-P-60	ALARA Strategy for Decommissioning Heavy Water Reactors	388
	Min Chul Kim, Gang Woo Ryu, Young II Na (KHNP-CRI)	
5-P-61	Analysis of Radiation Risk During PHWR Reactor Decommissioning	389
	Min Chul Kim, Gang Woo Ryu, Young II Na (KHNP-CRI)	

5-P-62	A Review on First Spent Fuel Disposal Facility: ONKALO Disposal Facility	390
	Young Hwan Hwang, Sunghoon Hong, Seong-Sik Shin, Seok-Ju Hwang, Jung-Kwon Son, Cheon-Woo Kim (KHNP-CRI)	
5-P-63	A Study on the Building Demolition Plan and Guidelines	391
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
5-P-64	A Study on the Safety Evaluation of Adjacent Units to be Considered in the Decommissioning of NPPs	392
	Young-II Na, Hyung-Woo Seo, Gang-Woo Ryu (KHNP-CRI)	
5-P-65	Review of the Media Considered at the Time of License Termination Phase of Overseas Decommissioning Nuclear Power Plants	393
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
5-P-66	Current Status of License Response to the Final Decommissioning Plan for Kori Unit 1 Related to Management of Decommissioned Radioactive Waste	394
	Suk Bon Yoon, Hyun Min Kim, Sukwon Jung, Gangwoo Ryu (KHNP-CRI)	
5-P-67	Development of Considerations for the Design of Wolsong Unit 1 Radioactive Waste Treatment Facility	395
	Gang Woo Ryu, Hyun Min Kim, Suk Won Jung, Suk Bon Yoon (KHNP-CRI)	

6분과 방사선환경 및 안전 [ORAL]

6-O-01	Tritium Monitoring to Monitor Ocean Discharge of TEPCO Treated Water	399
	Yukihisa Sanada, Hironori Funaki (JAEA)	
6-O-02	A Statistical Approach to Determine Evaluation Criteria for Proficiency Testing in the Environmental Radioactivity Analysis	400
	Cheol-Su Kim, Seong-A Yim (KINS)	
6-O-03	Determination of Effective Source Geometry for in Situ Gamma-Ray Spectrometry at Underwater Sediment	401
	Wanook Ji, Eunjoong Lee, Young-Yong Ji (KAERI), Kotaro Ochi, Sanada Yukihisa (JAEA)	
6-O-04	Airborne Radiation Survey in Fukushima and Analysis Algorithms	402
	Eunjoong Lee, Young-Yong Ji, Wanook Ji, Byoungil Jeon (KAERI)	
6-O-05	Proposal for the Design Criteria of Decommissioning Radiation-Controlled Area to Optimize Occupational Exposure in Decommissioning Nuclear Power Plants	403
	SeongJun Kim, Tae Young Kong (Chosun Univ.)	

6-O-06	Assessing Non-Radiological Risks of Lead in Gyeongju Radioactive Waste Disposal Facility	404
	Yulim Lee, Yeojin Kim, Ugyu Jeong, Jaeyeong Park (UNIST)	
6-O-07	Development of Sampling Device to Evaluate Radiation Deterioration Characteristic of NPP Bioshield Concrete and Environmental Analysis of Sampling Operations	405
	Dong-Yeon Kim, Sung-Hyeon Kim (NILEPLANT Co., Ltd.)	
6-O-08	Machine Learning Based Pre-Feasibility Test for Beta Nuclides Identification ..	406
	Min Ji Kim, Hee Reyoung Kim (UNIST)	
6-O-09	On-Site Demonstration of a Fast Compton/Coded-Aperture Hybrid Gamma Imaging System for Detection and Localization of Radioactive Materials	407
	Goeun Lee, Junyoung Lee, Taehyeon Eom, Dohyun Choi, Youngmo Ku, Chan Hyeong Kim (Hanyang Univ.)	

6분과 방사선환경 및 안전 [POSTER]

6-P-01	Analysis of the Criteria for Distinguishing Between Internal Exposure and External Contamination When Using Standing Type Whole Body Counter	411
	Suhyun Choi, Jongil Lee, Insu Chang (KAERI)	
6-P-02	A Method for Re-Evaluating the Release of Radioactive Materials Using Measured Radiation Doses After a Nuclear Power Plant Accident	412
	Si Young Kim, Jung-Kwon Son (KHNP-CRI)	
6-P-03	Feasibility Assessment of TLD Dose Algorithm Based on Artificial Neural Networks	413
	Soo Hyeok Lee, Hwi Joon Jeong, Kyung Taek Lim (Sejong Univ.), Hyungtaek Kim (KAERI)	
6-P-04	Review on the Method for Reduction of Radiation Dose by Radioactive Carbon in the Gaseous Effluent of NPPs	414
	Hang-Rae Cho, Seok-Ju Hwang, Si-Young Kim, Jung-Kwon Son (KHNP-CRI), Hyo-Jin Park (KHNP)	
6-P-05	A Study on Public Dose Assessment About Recycling of Radioactive Steel Waste Through Regulatory Clearance	415
	Kyungmin Kim (KAERI)	
6-P-06	Performance Test Results for Whole Body Counters in Nuclear Power Plants ..	416
	Siyoung Kim, Jung-Kwon Son (KHNP-CRI)	

6-P-07	Comparison Test of Reading Capability for Personal Dosimeters in Nuclear Power Plants	417
	Siyoung Kim, Jungkwon Son (KHNP-CRI)	
6-P-08	Changes According to the Application of the Cover Layer in Dose Evaluation for Site Release of Nuclear Power Plant	418
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
6-P-09	Estimation and Comparison of Worker Dose According to Decommissioning Schedule	419
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
6-P-10	Neural Network-Based Radiation Dose Calculator Program for Inhomogeneous Exposure Situations	420
	Yoomi Choi, Hyungtaek Kim, Sora Kim, Minchae Kim, Byung-II Min, Kyungsuk Suh (KAERI)	
6-P-11	Derivation of Time Response Curve of Radiation Survey Meter Using Cs-137 Standard Radiation Field	421
	Min Young Lee, Jun Beom Choi, Jang Jin Oh (KINS)	
6-P-12	Safety Monitoring and Strengthening Measures for Building Decommissioning Work	422
	Young-II Na, Gang-Woo Ryu, Hyung-Woo Seo (KHNP-CRI)	
6-P-13	International Studies on Extracting Dissolved Resources From Seawater	423
	Jieun Park, Wook Sohn, Kahee Jeong, Sooyeon Lim, Yujeong Shin (KHNP-CRI)	
6-P-14	Industrial Safety Management Cases for Decommissioning of Nuclear Facilities	424
	Jihyun Yu, Younjung Choi, Seonghun Kim, Jangsoo Suh, Hyerim Kim (Sae-An Enertech Corp.), Kwansong Jeong (KAERI)	
6-P-15	Analysis of Industrial Safety Accident Occurrences in Decommissioning Sites of Nuclear Facility	425
	Jihyun Yu, Seonghun Kim, Hyerim Kim, Jangsoo Suh, Younjung Choi (Sae-An Enertech Corp.), Kwansong Jeong (KAERI)	
6-P-16	Introduction of Environmental Radiation Investigation Method Around Nuclear Power Plant in Dismantled Nuclear Power Plant	426
	Byong Sop Lim, Hyu Chang Choi, Hyun Min Kim (KHNP-CRI)	
6-P-17	Conceptual Designs of the Automated Preconcentration and Purification System for Determination of Radionuclides in Seawater	427
	Kun Ho Chung, Wanook Ji, Uhee Jung, Young-Yong Ji (KAERI)	

6-P-18	Emitters' Contributions to Background Concentrations of Radioxenon in the Northern Hemisphere	428
	Kihyun Park, Byung-II Min, Sora Kim, Jiyeon Kim, Yoomi Choi, Jaemin Sohn, Kyung-Suk Suh (KAERI)	
6-P-19	Reflections on Radiation Protection Cooperation in Northeast Asia Based on the Analysis of the Cases of the Convention on Environmental Impact Assessment in a Transboundary Context Related to Nuclear Activities	429
	Heewon Kim (KINS)	
6-P-20	Development of Automatic Nuclide Analysis Capability for Airborne Radiation Monitoring	430
	Min Beom Heo, Jae Wook Kim, Hee Kwon Ku, Sang Hun Shin (FNC Technology Co., Ltd.), Hyun Jin Boo, Byung Gi Park (Soonchunhyang Univ.)	
6-P-21	Storm Surge Simulation Near Nuclear Power Plants Using ADCIRC and FVCOM Models	431
	Hyeon-Jeong Kim, Byung-II Min, Yoomi Choi, Ji Yoon Kim (KAERI)	
6-P-22	Prediction of Typhoon Path Occurring in the Northwest Pacific Using Artificial Intelligence	432
	Hyeon-Jeong Kim, Byung-II Min, Yoomi Choi, Ji Yoon Kim (KAERI)	
6-P-23	Wave Overtopping Simulation in Busan Marine City Using Smoothed Particle Hydrodynamics Model	433
	Hyeon-Jeong Kim, Byung-II Min, Yoomi Choi, Ji Yoon Kim (KAERI)	
6-P-24	Development of Domestic Accident Scenarios for Maritime Transportation of Spent Nuclear Fuel	434
	Kiwon Jang (KINS), Jaehyo Kim, Jooyub Lee (Seoul Natl. Univ.), Han-Ki Jang (KOARA)	
6-P-25	Construction of Carboxylic Acid Functionalized Covalent Organic Frameworks for Uranium Removal	435
	Bolam Kim, Muzammil Hussain, Dae Sung Lee (Kyungpook Natl. Univ.)	
6-P-26	Criticality Safety Evaluation of UO ₂ Power Storage for LEU+ Preparedness ...	436
	Sanggeol Jeong, Jang Hwan Lim, Seong Ki Lee (KEPCO NF)	
6-P-27	Optimizing Uranium Enrichment Analysis: A Comparative Study of ROI and Peak Fitting Techniques in Nuclear Safeguards	437
	Woojin Kim (KINAC)	
6-P-28	A Study on the Awareness of Physical Protection Among Nuclear Facility Employees	438
	Sang-Seong Kim (KINAC)	

Withdrawal

6-P-29	Enhancement of VR Content for Physical Protection Inspector Sang-Seong Kim (KINAC)	439
Withdrawal		
6-P-30	Highly Selective Cesium Adsorption From Acidic Solution Using Magnetic Calcined ZIF-67/Ammonium Molybdophosphate/Polyaniline Composites Youngsu Lim, Dae Sung Lee (Kyungpook Natl. Univ.)	440
6-P-31	Review on Overseas Case of Validity Verification of the Stack Sampling Technical Standard Hye Jin Jung, Jun Ki Baik, Se Youl Won, Dae Eun Park (KHNP-CRI)	441
6-P-32	Study of Transportation Risk Assessment Model Input Factor Sensitivity Evaluation Methodology Ga Eun Oh, Min Woo Kwak, Dong Gyu Lee, Kwang Pyo Kim (Kyung Hee Univ.)	442
6-P-33	Activation Analysis of ISOL Target Coolant in RAON Danhye Gil, Beomyeol Baek, Jongwook Kim, Kyungha Jo, Keunsoo Yang, Inseok Hong (IBS)	443

7분과 방사화학 [ORAL]

7-O-01	Synthesis of Uranium Chloride Through Hydriding and Chlorination of Uranium Metal for the Fuel Production of Molten Salt Reactor Dalsung Yoon, Sangkwon Lee, Seungwoo Paek, Ju Ho Lee, Taeho Kim, Chang Hwa Lee (KAERI)	447
7-O-02	Corrosion Behavior of Structural Materials for Molten Salt Reactor in NaCl-MgCl ₂ Salt Taeho Kim, Seol Kim, Eun-Young Choi, Dong Jun Shin, Seok Min Yoon, Chang Hwa Lee (KAERI)	448
7-O-03	Unary Molten Salt Synthesis of Tunable Adsorbents and Photocatalysts for Uranium Extraction From Seawater Shuang Liu (Seoul Natl. Univ.)	449
7-O-04	Range of Coefficients of Variation for Elemental Concentrations in Volatile Aerosols Generated From Molten Iron-Concrete Reaction Jei-Won Yeon, Minsik Kim (KAERI)	450
7-O-05	Determination of Radioactive Concentrations for Internal Structure of Reactor Vessel in Kori Nuclear Power Plant Unit 1 Myung Ho Lee, Byung Chul Song, Hee Seung Lim, Young Jae Maeng, Chang Hoon Lee (KRIST)	451

7-O-06	Consideration of the Standard (21 CFR 11) for Electronic Records and Electronic Signature in Radioactivity Analysis	452
	Seung-II Kim, Jung-Suk Oh, Chung-Sup Lim (KO-RA)	
7-O-07	Determination of ¹²⁹ Iodine in Cement Waste Form	453
	Gi Yong Kim, Jung-Weon Choi, Jung Bo Yoo, Tae-Hong Park (KAERI)	

7분과 방사화학 [POSTER]

7-P-01	Separation of Iron and Nickel in Radioactive Waste Sample by Cation Ion Exchange Chromatography	457
	Kwang Soon Choi, Seung-chul Park, Heejin Ahn, Mee Suk Song (Green Radiation Co., Ltd.), Jung Bo Yoo, Jai Il Park (KAERI)	
7-P-02	Properties of Uranium Nanoparticles for Solubility Test in Anoxic Groundwater	458
	Wansik Cha, Hye-Ryun Cho, Hee-Kyung Kim, Euo Chang Jung (KAERI)	
7-P-03	Investigation and Analysis of Cs-137/I-129 Nuclide Concentrations and MDA Data Characteristics for I-129 Nuclide in Radioactive Waste	459
	Sejun Pak, Soyeon Min, Dae Soo Lee, Gwang Tae Joo, Seung Jae Han (KORASOL Co., Ltd.)	
7-P-04	The Effect Evaluation Following as Amine Saturation Operation for Reducing Corrosion Products of Component Materials in Secondary System at NPPs	460
	Chorong Kim, Hyuk-Chul Kwon, Seung-Ho Lee, Daseul Ham, Yong-Sang Cho (KHNP-CRI)	
7-P-05	Study on Sample Pre-Treatment Methodology for Nuclide Analysis of Decommissioning Concrete Waste	461
	Jung-Weon Choi, Hyojin Ahn, Gi Yong Kim, Jung Bo Yoo (KAERI)	
7-P-06	A Study on Sample Drying Conditions for Soil Waste Gamma Spectroscopy	462
	Simon Oh, Kyunghun Jung, Jungbo Yoo (KAERI)	
7-P-07	Preliminary Study on Monte-Carlo Simulation of Gamma Detector for Low-Level Solid Radioactive Waste	463
	Kyunghun Jung, Simon Oh, Woojin Jo, Jung Bo Yoo (KAERI)	
7-P-08	The Safety Assessment About Generation of Radioactive Waste for Zinc Injection in Primary Coolant at NPPs	464
	Da-Seul Ham, Hyuk-Chul Kwon, Seung-Ho Lee, Cho-Rong Kim, Yong-Sang Cho (KHNP-CRI)	

7-P-09	The Consideration of Injecting Method for the First Zinc Application at AOA (Axial Offset Anomaly) Experienced NPPs	465
	YongSang Cho, KyuMin Song, HyukChul Kwon, SeungHo Lee, ChoRong Kim, DaSeul Ham (KHNP-CRI)	
7-P-10	The Effective Separation Method and Validation of Transuranic Elements (TRU) Such as Pu, Am, and Cm	466
	Myeongpil Kim, Changryong Lee, Heungnae Lee (KOSA)	
7-P-11	Accelerating Development of Structural Materials for Molten Salt Reactors: A Fast Screening Approach for Evaluating Corrosion Resistance	467
	Taehoon Park, Wonseok Yang, Jihun Kim, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
7-P-12	Development of Rapid Analysis Method for Chelates Contained in Radioactive Waste	468
	Han-byeol Kim, Sodam Jeon, Hye-young Kim, Taeyong Noh, Seoung-kyo Yoo (Wittech, Inc.)	
7-P-13	Effect of Oxygen and Moisture on Fission Product Release Behavior in the Operation of KCl- UCl_3 Molten Salt Reactor (MSR)	469
	Ju Ho Lee, Chang Hwa Lee (KAERI)	
7-P-14	Optimization of Pre-Treatment Conditions for Metal Waste Samples at KAERI	470
	Hyo Jin An, Jung Weon Choi, Jung Bo Yoo (KAERI)	
7-P-15	Performance Improvement Through Replacement of Chemical Hot Cell Manipulator	471
	Min-jae Ha, Hyung Kwon Lee, Hojin Lee, Hye-Ryun Cho, Jai Il Park (KAERI)	
7-P-16	Dissolution and Burnup Determination of Irradiated U-Mo Alloy Fuel by a Chemical Method Using Nd Isotope Monitors	472
	Hyejin Cho, Namuk Kim, Yang-Soon Park, Minjae Ha, Tae-Hong Park, Hye-Ryun Cho, Jai Il Park (KAERI)	
7-P-17	Improvement of Negative Pressure Measurement System in Chemical Hot Cell	473
	Min-jae Ha, Hyung Kwon Lee, Hye-Ryun Cho, Jai Il Park (KAERI)	
7-P-18	Feasibility Study of Radiochemistry Analysis for Radioactive Metallic Waste ..	474
	Sooyeon Lim, Kahee Jeong, Ji-eun Park, Wook Sohn (KHNP-CRI)	
7-P-19	A Review of ^3H and ^{14}C Analysis Methods	475
	Jieun Park, Wook Sohn, Kahee Jeong, Sooyeon Lim, Yujeong Shin (KHNP-CRI)	

7-P-20	Overseas Case Study: Applying the Mean Activity Concentration Method as an Indirect Evaluation for Radioactive Waste Isotopic Inventory	476
	Jieun Park, Yujeong Shin, Wook Sohn (KHNP-CRI), Seongjin Maeng, Jungi Yeom (KHNP)	
7-P-21	Construction of Radionuclide Analysis Laboratory as Test-bed for Decommissioning Wastes at KRID	477
	Seung-Ho Lee, Junghyun Park, Eunmi An, Homyung Lee, Seoha Kim, Yeong Su Ha (KRID)	
7-P-22	Establishment of a Quality Management System Plan for Rapid and Reliable Radioactivity Analysis of Decommissioning Waste at KRID	478
	Eunmi An, Yeong Su Ha, Junghyun Park, Seung-Ho Lee (KRID)	
7-P-23	Development of Pretreatment Technique for Analysis of Uranium in Simulated LiCl-KCl- UCl_3 Molten Salt Samples	479
	Chi-Gyu Lee, Ranhee Park, Chan-Yong Jung (KAERI)	
7-P-24	The Separation Method of Steel Samples by Ketone Separation	480
	YunHan Lee, DongGyu Lee, JunHwi Bang, Chang Ryong Lee, Heung Nae Lee, JeongSoo Kim (KOSA)	
7-P-25	Laser-Induced Breakdown Spectroscopy (LIBS) Measurements of Strontium Concentrations in FLiNaK Molten Salts	481
	Jihye Park, Hyeongbin Kim, Richard I. Foster, Sungyeol Choi (Seoul Natl. Univ.)	
7-P-26	Colorimetric Detection of Phenol and H_2O_2 by a New Peroxidase-Like Copper-Based Metal-Organic Framework	482
	Viktoriya Kim, Jeongmook Lee, Jong-Yun Kim, Sang Ho Lim (UST), Dong Woo Lee, Tae-Hyeong Kim (KAERI)	
7-P-27	Feasibility Study of Sample Amounts and Pretreatment Method for Spent Resin Sample Nuclides Analysis	483
	Kahee Jeong, Sooyeon Lim, Ji-eun Park, Wook Sohn (KHNP-CRI)	
7-P-28	Spatial Distribution of Metallic Precipitates in a Spent Fuel Assembly	484
	Junghwan Park, Sang Ho Lim (KAERI)	
7-P-29	The Separation Method of ^{55}Fe and ^{63}Ni by a Carbonate Precipitation	485
	DongGyu Lee, YunHan Lee, JunHwi Bang, Chang Ryong Lee, Heung Nae Lee, JeongSoo Kim (KOSA)	
7-P-30	Solubility Study of $\text{NpO}_2(\text{OH})(\text{am})$ in Oxidative KURT Groundwater Condition	486
	Hee-Kyung Kim, Hye-Ryun Cho, Sangki Cho, Wansik Cha, Tae-Hyeong Kim (KAERI)	

1분과

핵주기정책·규제 및 비확산 (ORAL)



Development of Research, Education, and Training Program on Nuclear Fuel Cycle and Economics Towards Advanced Reactor Technology at Virginia Commonwealth University

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With a \$1.5 million grant from the U.S. Department of Energy, Virginia Commonwealth University (VCU) is collaborating on research to advance the safety, efficiency, and environmental sustainability of nuclear energy. The funding will establish a pioneering research, education, and training center dedicated to advancing our understanding of the nuclear fuel cycle (NFC) and economic roadmap for next-generation reactor designs. The center will be bringing together experts from VCU and two historically black universities – Virginia Union University and Virginia State University – creating a one-of-a-kind educational and research training program that has never existed in Virginia, fostering diversity and collaboration.

The center's mission will be driven by five research objectives:

1. Unlocking the Economics of the NFC: Identifying key factors that determine the economic viability of different parts of the fuel cycle, from exploration and mining to enrichment and waste management. Understanding the economic tradeoffs and feasibility of each stage will be crucial in developing sustainable and cost-effective nuclear energy solutions.
2. Advanced Fuel Cycle Operations: Improving modeling and simulation of fuel cycle operations, giving a better understanding of processes involved. This will involve developing sophisticated computer models and experiments to optimize the performance and safety of nuclear reactors.
3. Designing the Future of Nuclear Fuel: Exploring cutting-edge concepts and technologies for reactor fuels, aiming for safety, reliability, and efficiency. Innovative fuel designs will be crucial in overcoming the challenges of current nuclear technologies, such as high costs and proliferation risks.
4. Revolutionizing Used Fuel Management: Developing advanced techniques and infrastructure for managing and disposing of used fuel safely and efficiently. This will include researching new methods for fuel reprocessing and waste treatment, as well as designing secure storage and transport systems.
5. Strengthening Nuclear Material Detection and Accountancy: Enhancing capabilities for detecting and quantifying nuclear materials, ensuring security and accountability. Improved detection technologies and systems will be essential in preventing nuclear proliferation and ensuring the peaceful use of nuclear energy.

These research objectives will be pursued across five core topic areas: materials science and chemistry, neutron transport, heat transfer and systems, nuclear material quantification, and engineering economics and finance. Each of these areas represents a critical component of the NFC and the development of next-generation reactors. By addressing the technical and economic challenges of nuclear energy, this research has the potential to transform the industry and contribute to a more sustainable future.

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Keywords: Nuclear fuel cycle, Economics, Material detection and accountancy, Nuclear materials

Tracking Footprints of Nuclear Security Centre of Excellence in the Republic of Korea

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This study highlights the achievement of the Nuclear Security Centre of Excellence (COE) in the Republic of Korea, which was initiated at the 2010 Nuclear Security Summit. The International Nuclear Nonproliferation and Security Academy (INSA), established in 2014 under the Korea Institute of Nuclear Nonproliferation and Control (KINAC), has been operating to advocate a cradle for capacity building on nuclear nonproliferation and security in Asia region. Quantitative records of INSA's education and training activities are summarised by dividing into three categories: (1) international training courses for neighbouring countries currently or potentially under nuclear cooperation and for newcomer countries for nuclear energy in peaceful use, (2) mandatory and compulsory education entrusted by relevant national law, and (3) awareness-raising to nuclear nonproliferation nuclear security for the public. For international training courses, the education operating system, including the development procedure of a new curriculum and the training of instructors for sustainability, is reviewed with remarkable achievement and circumstantial limitations. International cooperation, which is an integral part of the essentials of INSA, is addressed with the historical collaborating activities. In the process of implementing legal education in nuclear nonproliferation and security, the positioning of the KINAC/INSA as a technical support organisation to the nuclear regulatory authority is reviewed for strengthening and maintaining the national regime. Considering changes in the global environment and corresponding domestic changes, the importance of expanding awareness-raising activities against the public is emphasised. Further, the way forward from Asia to Asia-Pacific is presented with visions for the next five years.

Keywords: Nuclear security, Nuclear nonproliferation, Education and training, Centre of excellence (COE)

Investigating Improvement Strategies for Nuclear Nonproliferation and Security Awareness Programs in Domestic Human Resource Development

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In the year 2024, the Nuclear Safety and Security Commission has prioritized the development of a workforce endowed with a comprehensive safety mindset as a pivotal strategy for the establishment of a safety regulatory framework that aligns with international benchmarks. This strategic initiative aims to cultivate a cadre of professionals versed in the global norms pertaining to nuclear safety, security, and non-proliferation. The present scenario reveals a conspicuous shortfall in the provision of adequate educational programs within the realm of nuclear studies at both undergraduate and postgraduate levels in domestic institutions.

The Korea Institute of Nuclear Nonproliferation and Control (KINAC) has been at the forefront of executing educational endeavors designed to bolster the comprehension of nuclear nonproliferation and security amongst the nation's college and graduate student populations. These instructional sessions, which span an approximate duration of six hours, are meticulously planned and executed in collaboration with educational and institutional partners. Moreover, commencing from the year 2023, KINAC has embarked on the publication and dissemination of textbooks aimed at enhancing the familiarity of college students with nuclear nonproliferation and security issues. Despite these efforts, the existing short-term educational interventions exhibit limitations in fostering professionals with the requisite depth of expertise, underscoring the necessity for the formulation of specialized curricula that encapsulate comprehensive training, regulatory acumen, implementation strategies, and research methodologies.

This investigation undertakes a critical analysis of the prevailing programs aimed at augmenting awareness, with the objective of identifying and selecting pertinent course materials. Additionally, it conducts an evaluative review of nuclear control textbooks that have been distributed to institutions offering nuclear-related courses, with the goal of extracting and categorizing instructional modules into distinct themes of regulation, implementation, and research. Further, this study proposes the design of a specialized educational program tailored to enhance proficiency in nuclear nonproliferation and security regulatory compliance, leveraging the educational and practical training apparatus available through KINAC/INSA to optimize the educational outcome.

The educational paradigm introduced in this research is anticipated to significantly bolster the development of domestic nuclear nonproliferation and security personnel. By nurturing a workforce that conforms to international standards, this initiative is poised to contribute vitally to the realization of a nuclear nonproliferation and security regulatory framework that meets global expectations.

Keywords: Awareness programs, Human resource development, Nonproliferation, Security

Language Matters: A Corpus-based Analysis of the Korean Translation of ‘(High-Level) Radioactive Waste’ in Mainstream Korean News Media

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Considering the urgency of the topic, it is not surprising to see that ‘(high-level) radioactive waste’ has continuously been subject to news reporting. However, there seems to be no agreed corresponding Korean translation referring to the English term, ‘(high-level) radioactive waste’, resulting in multiple translations being used, even often interchangeably, from *kojunwibangsasöngp’yegimul* [high-level radioactive waste], *haekp’yegimul* [nuclear waste], *wönjaryökp’yegimul* [nuclear power waste] to *haekssüregi* [nuclear rubbish]. Each of these terms carries a different level of positive or negative semantic prosody, i.e. attitudinal meanings.

Against this backdrop, this study examines mainstream Korean news reports about ‘(high-level) radioactive waste’ with a view to identifying discursive discursal patterns constructed in *Chosun Ilbo* and *Hankyoreh*, the two Korean news media outlets representing centre-right and centre-left. Considering that news is socially constructed, and news media institutions would position themselves differently depending on an issue and their own ideological orientation, this study attempts to establish the relationship between textual features and practices specific to each news outlet. This study adopts Teun van Dijk’s definition of ‘discourse’ where social action and ideologies are represented; and employs a corpus-based method, a typically quantitative approach that allows the identification of typical linguistic behaviour or words in texts, in order to process large volumes of data and to achieve objective results. News reports between 2017 and 2022 held electronically were processed, and recurrent patterns were identified using a dedicated software program called KONLpy, a Python package for Korean natural language processing. The in-depth textual analysis has revealed *Hankyoreh*’s disproportionate use of ‘haek’ [nuclear]; and that ‘(high-level) radioactive waste’ is discussed in *Chosun Ilbo* in relation to international relations and policy, whereas it is considered as a domestic issue that requires social consensus.

Examining the two media institutions’ selective choice of a particular Korean translation over other possible translations (e.g. ‘haekp’yegimul’ over ‘kojunwibangsasöngp’yegimul’) for the English term (i.e. ‘(high-level) radioactive waste’) will give us the insight into the respective media institution’s positioning towards the high-level radioactive waste issue in Korea, and also will allow us to identify the two primary discourses about the issue in Korea, which is likely to influence the public. This ultimately will enable us to emphasise the coherent use of an agreed Korean translation of the term, and to develop a better communication strategy with the public for the high-level radioactive waste disposal site.

Acknowledgements

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Keywords: Corpus, High-level radioactive waste, News media, Framing, Discourse, Interdisciplinarity

Consideration of Nuclear Nonproliferation Technology Development Strategies for Advanced Nuclear Fuel Cycle

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The development of advanced nuclear reactor technology is in full swing in Korea, as well as in the United States and Europe. Whereas in the past, the development of nuclear system and its fuel cycle was led by government research, the recent development of advanced nuclear system and nuclear fuel cycle technologies linked to the development of SMRs has been actively participated in by industry. This is due to the lower barriers to entry for commercialization of advanced reactor types through SMRs, and such commercialization leads directly to exports. While the focus is on the development of advanced reactor system technology itself, it is clear that when these systems are eventually deployed domestically or exported overseas, they will need to be accompanied by a plan for how to manage spent nuclear fuel and how to supply nuclear fuel. Especially for countries without nuclear power experience, a domestic nuclear fuel cycle strategy linked to advanced reactors and solutions for international nuclear nonproliferation regulations should be provided as a package to ensure export competitiveness. In this study, the current status of nuclear cycle technology development was summarized and analyzed in order to consider a nuclear cycle model that can be linked to the advanced reactor system being developed by Korea. Specifically, future development strategies were derived based on the analysis of the current status of pyroprocessing and its safeguards technologies, which are representative examples of nuclear cycle technology development in ROK. The paper then extends this analysis to analyze nuclear nonproliferation issues for advanced nuclear systems and fuel cycles, and proposes a domestic technology development strategy for them. The current status and strategies summarized in this study are expected to be utilized in the establishment of mid- to long-term R&D strategies as well as in the ongoing international consultations on nuclear nonproliferation related to the development of advanced reactor technologies.

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Keywords: Nonproliferation, Safeguards, Advanced nuclear system, Nuclear fuel cycle, Strategy

Analysis of the Nuclear Materials Attractiveness Through New FOM Including the Nuclear Material Quantification Factor

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Nuclear material attractiveness is a methodology that evaluates utility in the production of nuclear weapons through the characteristics of nuclear materials. One of the representative methodologies for the nuclear materials attractiveness is the Figure of Merit (FOM). The FOM is a methodology developed by the Los Alamos National Lab that sets acceptance criteria for each factor, including critical mass, heat content, and dose rate, and evaluates the utility of nuclear materials in the production of nuclear weapons. In addition, the potential proliferation group's technological capabilities are being considered through spontaneous neutrons, which can cause a pre- detonation at an unwanted time.

As above, FOM is considering the nuclear materials attractiveness from the perspective of nuclear weapons production. Nevertheless, there are opinions that the FOM methodology lacks quantitative evaluation of nuclear materials. This is because the production of Pu according to the operating period cannot be considered. Therefore, this study added a quantitative factor for nuclear materials to the existing FOM and analyzed how the attractiveness of nuclear materials changes depending on the quantity of nuclear materials.

As a method for analyzing this, first, a pressurized light water reactor was modeled through Monte Carlo N-Particle (MCNP) to calculate the production of Pu according to the operating period of the reactor. Second, the FOM was calculated through MCNP and ORIGEN-S, and as a criterion related to the quantification of nuclear materials, 1 Significant Quantity (SQ), which is a non-negligible amount for production nuclear weapons, was used. The name of the FOM newly suggested and analyzed in this study was called to FOM_Q.

As a result of this study, the amount of Pu generated per assembly when operating for 12 months is about 2 kg. In this case, FOM1 is 2.63, and the value of FOM1_Q is 0.4. If this is expanded to the entire core, the amount of Pu is about 350 kg, and the FOM1_Q value at this time is 2.34. Even for the same operating period, when the mass of nuclear material is little, the nuclear material attractiveness is 'unattractive', but when the mass increases, the nuclear material attractiveness rises to 'preferred'.

Through the results of this study, it was confirmed that nuclear materials attractiveness increased with mass. This suggests that utilizing the existing FOM methodology allows for analyzing changes in attractiveness based on the quantity of nuclear materials. In the future, the reliability of this methodology will be analyzed through correlation analysis between newly developed FOM and mass.

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Keywords: Nuclear material attractiveness, Figure-of-merit (FOM), Non-proliferation resistance, Quantitative factor

1분과

핵주기정책·규제 및 비확산 (POSTER)



Implementation of a Nuclear Regulatory Law Search System Using the RAG (Retrieval Augmented Generation) Model

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As large language model (LLM) develops, starting with OpenAI's Chat-GPT, groundbreaking changes are occurring in interactive information retrieval. American big tech companies, such as Microsoft, Meta, and Google, are releasing models that have been trained on their own data. These models produce high-quality answers in common questions but may not provide correct answers in specific areas. Two methods are generally used to improve response performance in specialized fields.

The first method is to fine tune foundation LLM by building a refined learning dataset. Although a lot of computing resources must be invested in learning, creative and specific answers can be expected. The second method is to use a model that links RAG model and LLM. The RAG model stores learning data as a vector database using an embedding model, and extracts vector data with high similarity to the user's question. And, it generates an appropriate prompt combined with the user's question to generate the answer entered into the LLM. Although RAG model is difficult to expect answers that understand the context of the question, the source is clear and hallucinations are reduced. Additionally, even if training data is added or changed, model update can be performed only through vector embedding rather than LLM retraining.

In order to use the LLM in a nuclear regulatory law search system, the accuracy of the answer is more important than understanding the context and it is necessary to respond to frequent law amendments. Therefore, in this paper, we established a RAG system and tested answers to related questions using embedded nuclear energy regulatory law documents. As a result of analyzing the answers, the model generated appropriate answers and it appears that it can be used in the knowledge management system by adding related technical data in the future.

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Keywords: Nuclear regulatory law, Large language model, Retrieval augmented generation

Gap Analysis of Regulatory Standards for Applying Risk-Informed and Performance-based Regulation in Nuclear Cybersecurity of Republic of Korea

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There is a growing recognition of the need for the introduction of Risk-informed and Performance-based (RIPB) regulations for domestic nuclear facilities in order to efficiently utilize limited regulatory resources and to enhance nuclear operator's voluntary cybersecurity improvements. The key features of RIPB regulation is that this regulation focuses regulatory efforts on facilities with higher risks and degraded safety performance.

The U.S. Nuclear Regulatory Commission (NRC) has been implementing regulations based on RIPB approaches, using the Reactor Oversight Process (ROP) as a framework for regulatory oversight. In the Republic of Korea (ROK), various studies have been conducted to introduce RIPB regulations based on the ROP framework of the U.S. NRC. Specifically, in the safety regulatory domain in the ROK, the research and development of RIPB safety regulatory measures based on the U.S. ROP have been progressed, and its results have been tried out. However, in the security regulatory domain in the ROK, only the need for applying the RIPB on security regulatory domain has been raised, and currently, the Korea Institute of Nuclear Nonproliferation and Control (KINAC), the delegated agency for nuclear facility cybersecurity regulation, has been researching on the development of security-related RIPB regulations.

In the ROK, nuclear security regulations are implemented based on the "Act on Physical Protection and Radiological Emergency (APPRE)" with detailed provisions following subordinate regulations, guidelines, and procedures from the Nuclear Safety and Security Commission (NSSC), the regulatory body, and KINAC. However, the current nuclear security regulations in the ROK do not apply RIPB regulations. Consequently, it is anticipated that when applying RIPB regulations on nuclear cybersecurity of the ROK in the future, there will be some limitations such as insufficient legal basis for regulatory activity and lack of regulatory consistency due to the absence of regulatory standards.

In this study, we derived approaches for introducing RIPB regulations on nuclear cybersecurity in the ROK and conducted gap analysis of cybersecurity regulatory standards for implementing the RIPB security regulation in the future. Furthermore, based on the analyzed results, we derived necessary regulatory criteria for the rule-making and amendments of RIPB security regulations within the scope of the existing security regulatory standards, including APPRE and NSSC guidelines and inspection criteria.

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Keywords: Reactor oversight process, Regulatory structure, Nuclear security, Cybersecurity, Risk-informed performance-based

Improvements in Nuclear Export and Import Control System (NEPS) Functionality and Future Enhancement Directions

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The Nuclear Safety and Security Commission (NSSC) and the Korea Institute of Nuclear Nonproliferation and Control (KINAC) operate the Nuclear Export and Import Control System (NEPS), which handles civil complaints and manages information related to the export and import of trigger list items. Efforts to expand operational functionality and improve convenience are made annually, including a comprehensive system rebuild in 2020. This paper examines three significant NEPS functionality improvements made in 2023, aiming to outline future enhancement directions.

First, we developed a function to handle tasks related to the Nuclear Cooperation Agreement with the UAE. In January 2023, the Administrative Arrangement of the ROK-UAE Nuclear Cooperation Agreement was signed, and related functions were developed and added to NEPS to facilitate the international transfer information exchange procedures (prior notification and its acknowledgment, shipment notification, receipt confirmation, and the like) between the two countries. As a result, 15 ROK-UAE information exchange procedures were efficiently processed using the system within a year.

Secondly, the process for advance notification of international nuclear material transfers was enhanced. The original process required users to enter application information directly into the online system and then re-enter the same information into an English format document for additional attachment, leading to unnecessary administrative effort and data inconsistency issues. In response, the content of the application form for advance notification of international transfers was modified, and the process was improved to only require entry into the online system. Additionally, features were added to generate official letters for IAEA reporting. These strengthened the system for timely fulfillment of international obligations and improved user convenience.

Lastly, the functionality and menu structure of the NEPS subsystems were optimized. Optimizations for the civil complaint system, review system, and mobile system within NEPS included search menu and grid item rearrangements for enhanced search capabilities, menu restructuring through integration, separation, or deletion of menus, removal of unnecessary duplicated features across systems, and interface improvements. These efforts increased efficiency and consistency across systems and improved convenience and operational efficiency for users and reviewers.

Due to the activation of various nuclear-related export businesses, there are plans to improve NEPS continuously. In response, continuous improvements to NEPS are planned. Plans include integrating the currently separate civil complaint system and portal systems for better user accessibility and enhancing the review function. Additionally, there is a commitment to continually incorporate user feedback to enhance the convenience and efficiency of civil complaint processing.

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Keywords: Nuclear export, Nuclear import, NEPS, Feature enhancement, Functionality

Analysis on the Operation of the Nuclear Pre-Consultation System

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After the construction project of the UAE Barakah Nuclear Power Plant (BNPP), several contracts have been secured, and recently signed the supply of equipment for a nuclear power plant in Egypt and the construction of a tritium removal facility in Romania. In addition, there are expectations for securing contracts for the construction of nuclear power plants in the Czech Republic and Poland.

In order to carry out nuclear-related export projects, each company must implement export controls in the case of transferring items or technologies from submitting bids to the pre-project stage. Additionally, as the list of items subject to the Russian sanctions has recently been expanded, it is necessary to conduct a detailed review for export or catch-all license for items planned to be exported to Russia.

However, in some cases, the timeline for each stage of the export project is tight, or classification and export licenses are applied for close to the export project schedule. Therefore, the NSSC and KINAC have established the nuclear pre-consultation system from 2023 to check the possibility of issuing export licenses in advance and provide technical support for the smooth implementation of export controls in order to comply with the export project schedule. The consulting program is aimed at domestic companies promoting nuclear related export (1) to check the compliance with export control requirements in advance, (2) establish a customized export control implementation plan that reflects the characteristics of its business conditions, and (3) provide detailed guidance on export control implementation.

From March 2023 to March 2024, a total of 33 consultations were conducted, indicating a continuous increase in demand for pre-consultation. The object of pre-consultation were 13 items, 1 software, 17 technical data, and 2 technical support. 20 cases were Trigger-List items (TL) and 13 cases were non-TLs. 7 cases among 20 TLs required Government Assurance and 13 cases were identified as either following Nuclear Cooperation Agreement's transfer procedure, or being exportable to nuclear-weapon states without Government Assurance.

The pre-consultation results enabled businesses to implement export controls quickly and effectively. Regulatory agencies could simplify administrative procedures and enhance work efficiency by identifying necessary information in advance. As nuclear-related export business is actively progressing, the government and regulatory agencies will continue to make more efforts to support overseas export projects, such as the introduction of the pre-consultation system.

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Keywords: Nuclear export, Export control, Consultation system

Review of the Necessity for a Self-Inspection System in Post-Management Obligations of Nuclear Power Plant Technology Export License

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The Nuclear Plant Technology Export License System was introduced in 2014, allowing for the consolidated approval of technology exports throughout the project duration. This system was implemented to faithfully comply with the international regulations of the multilateral export control regime while enhancing administrative efficiency and establishing a systematic management system through project-specific export licenses. Through this licensing, post-management is being conducted for 12 projects, including 7 UAE projects such as the BNPP Operating Support Services Agreement, as well as 5 other projects including the Bangladesh Research Reactor Improvement project.

The license system targets technical documents such as design drawings, manuals or, procedures, and technical support including personnel dispatch. Post-management obligations include quarterly reporting of transfer history, submission of export control training course certificates for dispatched personnel, and submission of individual export control pledge documents.

Since 2021, the government and regulatory agencies have been conducting export control inspections to check the implementation status of licensees to identify improvements and assess the appropriateness of licenses. However, due to the limitations of time and personnel to inspect all projects of various licensees, the introduction of a self-inspection system for licensees to check their own post-management obligations is necessary.

The Self-inspection contents are divided into export control implementation systems, contract management status, transferred technical document management status, and transferred technical support management status. In the internal export control implementation system, the licensee's relevant organization and personnel status are checked, internal regulations related to export control and the management system for export control of trigger list items are checked. For contract management status, the history of contract changes and applications for export license modifications will be inspected, as well as subcontracting agreements and their compliance with export control. The technical document and support management statuses will also be inspected to verify the appropriateness of export control implementation. In addition, the self-inspection will be provided in a descriptive format with examples to allow licensees to identify and rectify deficiencies on their own.

The government and regulatory agencies plan to require a preliminary self-inspection for post-management obligations to licensees in 2024 to reflect necessary improvements to the self-inspection, and to establish a self-inspection system for licensees from 2025. It is anticipated that licensees will be able to thoroughly conduct post-management obligations by identifying improvement measures by themselves, and regulatory agencies may omit frequent inspections by verifying the checklist of self-inspections and the reflection of improvements.

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Keywords: Nuclear export, Export control, Export control inspection, Self-inspection

Regulatory Improvements for Safeguards Implementation From the Design Phase

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The Safeguards by Design (SBD) initiative is championed by the International Atomic Energy Agency (IAEA) to incorporate safeguards into the design and development of new nuclear facilities from the beginning. This initiative is crucial in ensuring that safeguards measures are considered preemptively. The IAEA's Member State Support Programme (MSSP) plays a crucial role in facilitating international cooperation, in particular with the Republic of Korea (ROK), among others, to tailor and implement safeguards measures explicitly designed for small modular reactors (SMRs). The ROK's participation in the 'Safeguards by Design for Small Modular Reactors' project through MSSP demonstrates its support for strengthening nuclear safeguards. In the project, the IAEA required the voluntary submission of the Design Information Questionnaire (DIQ) and the Safeguards Technical Report (STR) for new nuclear facilities. These documents are essential for incorporating safeguards measures from the design phase onwards for the SBD by the Agency.

However, discrepancies in the current regulations in the ROK, such as the voluntary submission of the DIQ to the IAEA in advance, highlight the need for regulatory amendments. This study proposes harmonizing and synchronizing of DIQ submission protocols with licensing processes to ensure early implementation of safeguards in the ROK. Additionally, it suggests creating guidance for helping ROK designers on collaborating with the IAEA for STR development. The study's results can be used effectively and practically to amend the regulations by including the draft recommendations on regulatory documents.

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Keywords: International atomic energy agency (IAEA), Nuclear safety act, Safeguards, Regulatory amendments, Design information questionnaire (DIQ)

Comparison of Policies During Trump and Biden Administration: Focusing on Nuclear Nonproliferation Issues

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Results of the 2024 Super Tuesday confirmed a re-match between current U.S. President Joe Biden and former president Donald Trump for the 2024 Presidential Election. On the surface, the two candidates seem to have vastly opposing ideologies as well as political positions. During his presidency as well as the 2020 and 2024 presidential campaign, Trump has consistently emphasized his goal to “Make America Great Again (MAGA)”. Meanwhile, Biden placed more focus on a global pro-democracy achieved through strengthening cooperation between U.S.’s allies and partners. Despite the stark differences between the two candidates, this paper argues that both candidates have maintained a similar stance of “America first” approach in conducting domestic and foreign policy during their first term as president. This is because the decline of manufacturing industries in the Rust Belt, fear of U.S. workers being replaced by immigrants, and concern over becoming entrenched in allies’ conflicts abroad made it difficult for any presidential candidate to be elected without addressing these issues. The objective of this paper is to identify which issues remained unchanged regardless of president personality and Party preference. More specifically, this paper seeks to identify the status of nuclear energy policy during both Trump and Biden administration and whether this will change based on the 2024 presidential election. To do so, this paper will first compare Trump and Biden administration’s policies. Second, it will focus on each administration’s nuclear nonproliferation policies to identify if there was policy continuation or reversal. Finally, the paper will conclude with a tentative forecast that U.S. policy trends will not drastically change during either Biden or Trump’s second term.

Keywords: U.S. election, Trump, Biden, Nuclear nonproliferation, Nuclear energy

Analysis of SMR Export Business Status in Korea and Improvements for SMR Export Control

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Recently, domestic nuclear companies have been actively promoting global SMR business based on their accumulated experience in the production, construction, and operation of NPP. The cases of SMR export are divided into the export of Korean-type SMR (i.e. SMART, i-SMR) and the support for overseas SMR (i.e. Nuscale, Rolls-Royce, etc.) projects. This study analyzed the status of SMR export business in Korea by classifying it into five categories: 1) exporting Korean-type SMR, 2) transferring technology to receive technical advices on Korean-type SMR, 3) outsourcing parts of SMR business to foreign companies, 4) supplying components of overseas SMR model, 5) providing technical support for overseas SMR model.

Consequently, the characteristic of the SMR export business and improvements for SMR export control have been identified as follows. Firstly, SMRs are applying various types of reactor. Therefore, design features of each type of SMR needs to be comprehensively reviewed from an export control perspective. Secondly, most SMR models are still under research and development, leading to active technology transfer through international cooperation, rather than the transfer of goods. Therefore, efficient export controls measures need to be established focusing on technology transfer. Thirdly, the participation of start-ups and small companies is expected to increase, as the SMR business is relatively small and the SMR model is still in the development phase. Outreach activities targeted to start-ups and small companies are required by monitoring the participating companies continuously.

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Keywords: SMR, Export control, International cooperation, Technology transfer, Outreach

A Study on How to Improve Nuclear Export Control Outreach Program

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Korea Institute of Nuclear Nonproliferation and Control (KINAC) conducts various outreach activities to improve regulated parties' understanding of nuclear export controls and reduce regulatory blind spots. Recently, nuclear-related exports, including nuclear power plants, have increased, and various domestic and international policies have changed. Consequently, KINAC is working on a plan to make export control outreach more efficient and effective. To this end, an analysis was conducted on the trends in nuclear export controls, the outreach activities of other export control organizations, and the current status of KINAC's outreach. Based on the results of these analyses, the following improvements were identified to improve outreach.

First, customized outreach considering the characteristics of each company and project is needed. New large-scale nuclear power plant projects in the Czech Republic, Poland, and other countries, as well as export projects such as SMRs, are being activated in various ways. Traditional group training faces limitations in addressing the increasingly diverse individual projects. Thus, outreach that considers each project's specific features is increasingly required.

Second, outreach should be more aligned with export control implementation. In many cases, outreach programs for regulated entities or potential regulated entities, such as institutional guidance and education, are not linked to export control implementation. In addition to offering guidance and education, it's crucial to enhance the connection to export control implementation through detailed technical support, such as verifying the possibility of obtaining export licenses at the start of a project.

Third, efficient use of limited outreach personnel and budget is necessary. With the surge in export control issues, such as sanctions against Russia, there is a shortage of workforce and budget for outreach. Therefore, it is necessary to select activities from existing outreach efforts that can be transferred to other departments to save resources. Additionally, considering integrated outreach by regionally grouping companies can also increase efficiency.

The following detailed implementation strategies were prepared to reflect the improvement measures identified. A new system of 'Nuclear Export Control Pre-Consultation' will be implemented to reflect the project-specific characteristics and enhance the linkage with the implementation of export control. Pre-consultation is one of the critical priorities of the Nuclear Safety and Security Commission, and customized technical support will be provided to ensure the smooth implementation of export controls. Existing outreach activities targeting universities will be merged into educational training conducted by the Education and Training Division. Based on this, efforts will continue to perform sustainable and efficient outreach to enhance regulatory parties' understanding and reduce regulatory blind spots.

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Keywords: Export control outreach, Regulatory compliance, Nuclear export, Customized outreach

Research on Security Regulatory Documentation Between the Domestic and NuScale Case

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More than 80 Small Modular Reactors (SMRs) are under development worldwide, such as the U.S., Canada, Russia, the U.K., China, and others. According to the development of SMRs, nuclear regulatory bodies in the U.S., U.K., and Canada have established close cooperation with SMR developers and are proceeding with the review process.

Currently, the Republic of Korea (ROK) is in the process of developing an innovative SMR (i-SMR) capable of producing up to 170 MW(e). The major milestones include finalizing the basic design and standard design by 2026 and obtaining regulatory certification by 2028. Due to the different design aspects of i-SMR, the applicability to existing security requirements should be analyzed, and modifications may be required. Therefore, security regulatory strategy for i-SMR are necessary, such as the documentation, requirement, methodology based on safety and security integration.

This study aims to analyze the security regulatory gaps based on the NuScale case by U.S. NRC in order to develop security regulatory documentation and to utilize them as references. Firstly, security regulatory documentation developed and applied for NuScale is analyzed. During the pre-application reviews, the Design-Specific Review Standard (DSRS) for NuScale Design has been developed to provide guidance to the Nuclear Regulatory Commission (NRC) technical staff for reviews not covered by the existing NRC standard review plan (SRP). The NRC has initiated a rulemaking process referred to as Title 10 of the Code of Federal Regulations (CFR) Part 53 and Part 73.110, aimed at suggesting a new regulatory framework for advanced reactors along with accompanying regulatory guidance. Secondly, security regulatory documentation applicable to i-SMR is reviewed. For cyber security of domestic nuclear facilities, KINAC/RS-015 is applied as a requirement under the 「Act on Physical Protection and Radiological Emergency」. Compared to the regulatory documentation applied to NuScale, practical regulatory documentation for security regulation of i-SMR remain at an early stage. Therefore, it is necessary to develop regulatory documentation that organizes security concerns based on technical reports submitted by licensees during the pre-design review process. Then, it is necessary to establish detailed review guidelines specified to the i-SMR design, adapting the existing security regulatory requirements. It should be utilized for upcoming standard design certification. In domestic law, the design certification is not integrated to the security review, so the safety-security integrated review is necessary. Furthermore, it may be necessary to publish security regulatory documents describing lessons learned during the review process.

Keywords: SMRs, Security, Regulation, Documentation, NuScale

FOM-based Nuclear Proliferation Assessment on the Application of HALEU in ATF

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High-assay low-enriched uranium (HALEU) is a nuclear fuel with a U-235 enrichment of 5-20%, and is interesting as a fuel with the potential advantage of reducing the volume of waste and securing economic efficiency by increasing the fuel cycle length. Contrary to this, the higher enrichment level of HALEU means less work is required to turn it into weapons-grade HEU. Therefore, it shows a weakness in terms of nuclear proliferation.

Based on above observation, we decided to study on the nuclear proliferation of HALEU because it is required to investigate the change of risk in terms of nuclear proliferation when HALEU was depleted as a function of time.

Recently, there was a study that analyzed attractiveness by applying Accident tolerant fuel (ATF) to SMR. In that study, U₃Si₂ fuel was proven to be superior to UO₂ fuel in basic neutronics, safety, and security characteristics, therefore, U₃Si₂ was selected as a fuel to be applied in this study. The target reactor type for this study is SMART (System-integrated Modular Advanced Reactor), a small-sized integral type PWR, and the nuclear fuel enrichment is 2.82wt% and 4.88wt% of U-235. We measured nuclear proliferation at the point of 990 EFPD in three cases: when using existing LEU nuclear fuel and when using HALEU with 7.82wt% and 9.88wt% (add 5% to existing fuel) and when using HALEU with 12.82wt% and 14.88wt% (add 10% to existing fuel).

For the evaluation, the Figure of Merit (FOM) methodology was applied. FOM is a methodology for evaluating the attractiveness of nuclear materials, which consists of parameters such as critical mass, dose, and heat generation rate for nuclear proliferation evaluation. Currently, depletion simulation for U₃Si₂ pellets was performed based on the SMART design parameters using the MCNP 6.2 (Monte Carlo Neutron-Particle) code. As a next step, the SCALE 6.3 code will also be used to obtain the heat generation rate and raw data to evaluate the dose rate. Finally, we plan to evaluate and compare the FOM values between LEU-type spent nuclear fuel and HALEU-type spent nuclear fuel.

Despite the interest and concerns about ATF and HALEU, data on them are lacking compared to existing fuels. This study focused on studying the change in ATF characteristics from a nuclear proliferation perspective when the fuel enrichment level in SMR was increased to the HALEU level. Through this study, according to the fuel U-235 concentration, risk changes of nuclear proliferation can be used as basic research data to identify changes in risk factors in terms of nuclear proliferation due to the expansion of HALEU use.

Keywords: Figure-of-merit (FOM), Accident-tolerant fuel (ATF), Small modular reactors (SMR), High-assay low-enriched uranium (HALEU), Non-proliferation

Measurement of LiCl-KCl Salt on the Dipstick Using FO LIBS and Spectral-based Normalization Methods for the Quantitative Analysis

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In-situ measurement of LiCl-KCl salt plays important role in the pyroprocessing safeguards. A Fiber-Optic (FO) Laser Induced Breakdown Spectroscopy (LIBS) system was employed to analyze LiCl-KCl salt deposited on a dipstick. The laser beam was delivered through one optical fiber, and the light emitted from the plasma generated on the salt surface was transmitted through another optical fiber. The measured LIBS spectrum covered a wavelength range of 450-650 nm, which was chosen considering the radiation effect on the optical fiber. The spectral peaks corresponding to elements in the salt such as Li and Fe were identified. The intensity of the Fe peak was measured along the dipstick, and its variation was consistent with the observed change in the Fe composition. However, the LIBS peak intensity is highly sensitive to the distance between the LIBS probe and the salt surface. To minimize this for the quantitative analysis, different spectral-based normalization methods were evaluated. LIBS spectra of a LiCl-KCl salt sample containing Nd, and a stainless steel sample were acquired varying the distance between LIBS probe and sample distance (0.5-4.5 mm). Normalization methods such as background, total area, internal standard, and standard normal variate (SNV) were applied to minimize the impact of distance on peak intensity. Only the internal standard methods were effective for the stainless steel sample, while internal standard, total area, and SNV were all effective for the LiCl-KCl sample.

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Keywords: LIBS, Safeguards, Pyroprocessing, Normalization

Feasibility Study on Random Number-based MBE Method for MUF Uncertainty in Benchmark Bulk Handling Facility

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Since adopting the State Level Approach of the International Atomic Energy Agency (IAEA) in September 2015, South Korea has conducted independent national safeguards inspections for domestic nuclear facilities. The IAEA defines the material accounting as a procedure to verify whether the declared information and physical inventory of nuclear materials.

The IAEA has employed a material balance evaluation (MBE) to evaluate the facilities' information on nuclear material accounting. The MBE evaluates whether the material balance (i.e. material unaccounted for (MUF)) within a material balance period comes from measurement uncertainty. The IAEA adopted deterministic approach, which estimates the combined uncertainty using the Taylor series, for batch-based individual measurement uncertainty. However, it does not consider the characteristics of accounting systems of individual facilities and the accumulation of estimation can cause bias in the combined uncertainty. Therefore, this study proposes new approach applying Monte Carlo techniques to evaluate the uncertainty in MUF and aims to assess its validation through comparison with the existing IAEA method.

In prior research, the IAEA method was applied to an arbitrary list of inventory items, with ^{235}U MUF set to 0 kg, yielding a measurement uncertainty for MUF (σ_{MUF}) of 37.578 kg. In this study, using MATLAB, normal distribution random numbers for mass, sampling, gravimetry, and isotope ratio measurements were generated with a mean of 1 and relative standard deviation of the international target values (ITV) uncertainty corresponding to the four measurement methods. These random numbers were multiplied to each ^{235}U weight, and the material balance quantity of these pseudo-measured values was calculated to derive the MUF value. Subsequently, after considering calculation time, approximately 500 iterations were performed, and σ_{MUF} was calculated.

The uncertainty for 500 iterations showed a difference of approximately $\pm 0.5\%$ compared to the σ_{MUF} by the IAEA method. Furthermore, performing t-tests between the calculated σ_{MUF} from 30 repetitive calculations and the σ_{MUF} of the IAEA method resulted in a p -value of 0.38, indicating no significant difference between the random number method and the IAEA method. It demonstrated that the results obtained from the stochastic method were comparable to those obtained from the deterministic method in material accounting for bulk handling facilities.

In future research, the optimal number of iterations considering calculation time and MUF uncertainty will be determined, and comparisons will be made with results based on the guide to the expression of uncertainty in measurement (GUM) for MBE, in addition to the existing IAEA method.

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Keywords: MBE, MUF, Measurement uncertainty, Nuclear materials accounting, Pseudo random number

A Study on the Improvement and Optimization of KAERI Export and Import Control System

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KAERI fulfills its export and import obligations for strategic items in accordance with domestic laws. Strategic items are goods, software, and technologies available for the manufacture, development, use, or storage of weapons of mass destruction and their means of transport, missiles. The purpose of export and import controls is to prevent or block strategic items in a particular area from entering the hands of non-state actors as terrorist group. This is called Nonproliferation Control.

For non-proliferation control, ROK implements its obligations to control exports and imports by complying with the notification of export and import of strategic items in accordance with the Nuclear Safety Act and the Foreign Trade Act. Nuclear export control items within the strategic items export and import notification are classified to Trigger List items and Dual-use items.

KAERI has complied with domestic law and implements the export and import control obligations. To this end, a KAERI'S Export and Import control system was established to efficiently perform the import and export obligations within the institute in 2020. This paper will deal with the current status of the Export and Import control system established within KAERI and the need for continuous improvement.

In 2021, KAERI's Nuclear Safeguards and Export Control Section introduced the new export and import control system for smooth application and management. It is KAERI Export and import Control System (KECS). This system is export and import control 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.

In order to meet the user's requirements, it is necessary to minimize work delays through maintenance operations and response to errors. In addition, KECS aims to maintain consistency across the environment and reduce the workload of practitioners by systemizing daily management tasks. Through this system improvement, the following advantages can be obtained. The benefits include reduced risk in terms of security and compliance, and freeing up time for strategic planning.

The number of license Application on Technology Export of Nuclear Plants are expected to increase further in the future, and follow-up management through quarterly reports is expected to be systematic. It is expected that the number of export permits within the Atomic Energy Research Institute will further increase in the future. We will continue to make improvements so that the simplification of work for applicants and practitioners can be realized within KECS.

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Keywords: Export and import control, KECS (KAERI export and import control system), Strategic items, Trigger list, Dual-use

Considerations for Adopting the Obligation Code for Domestic Management of Imported Internationally Controlled Materials

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When importing Trigger List Items (TL) or Items subject to bilateral nuclear cooperation agreements (ISA), the supplier requires a written guarantee from the ROK for peaceful use, safeguard applications, and prior consent for re-transfer. The obligation to control these internationally controlled materials (ICM) persists nationally post-import. Currently, importers track these based on origin, complicating adherence to agreements. To address this, KINAC is creating an Obligation Tracking System (OTS) for ISA, proposing an obligation code system distinct from origin-based management. This study explores assigning obligation codes and considerations for expanding the system to include TL imported under government-to-government assurance (GTGA).

To manage obligations for imported ICM, especially nuclear materials, some countries use ‘obligation codes’ of one or two letters or numbers, usually assigned on a ‘country’ basis. Canada applies an alphabetic country code, while the U.S. uses a numeric system. U.K. further assigns codes indicating obligations for peaceful use (P) or safeguards (N), enhancing the specificity of control beyond just country identification.

The OTS currently under development focused on ISA with five countries; therefore, country-based obligation codes can be used as standard practice. However, the following points should be considered for future expansion to include government-guaranteed products. First, the code should adopt an efficient way to distinguish countries in the world. For equipment, a two-digit country code could be applied, as in the case of Canada. In contrast, listing all relevant countries for managing obligations tied to conversion and enrichment is impractical for nuclear materials. Therefore, it is necessary to review the technology related to the nuclear cycle and TL each country possesses to reduce the number of cases where it needs to be managed as a significant supplier.

Second, a categorization based on “obligations”, similar to the U.K.’s approach, is suggested. Because even the items imported from the same country may need to be controlled differently depending on their obligations. For example, the U.S. requires GTGA for TL, which is not part of the ROK-U.S. NCA. This necessitates codifying selective combinations of some requirements, such as dividing how re-transfer is controlled or how assurance is provided.

Third, the form of the code should be considered to effectively and efficiently communicate country and mandate information. The obligation code must effectively convey essential information between regulators and domestic ICM users, factoring in the system’s computerized management for its form and length. It could utilize an English country code and a numerical list.

By reviewing practices from other countries, this study proposes a framework that combines country-based and obligation-based codes to ensure compliance with bilateral agreements and GTGA. Considering the efficient form when expanding an obligation code system will help to systematize the management of ICM and enhance the adherence to international obligations.

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Keywords: Export control, Nuclear cooperation agreement, Obligation code, Nuclear material, Trigger list item

Development of Nuclear Proliferation Risk Assessment Methodology Based on Data Analysis and Deep-Learning Technologies Using Open-Source Information

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The export of the UAE Barakah nuclear power plants in 2009 meant that Korea had become a nuclear supplier country in the international nuclear non-proliferation regime and regulation. Therefore, while fulfilling its responsibilities and obligations regarding nuclear non-proliferation, Korea also needs to exert stronger leadership in the field of nuclear non-proliferation, particularly in Northeast Asia, which is a region where the interests of great powers (e.g., the United States, Russia, and China) are intricately intertwined and where various elements of conflict are inherent. To make matters worse, nuclear security concerns have been raised in recent years owing to the continued nuclear activities of neighboring countries. To prepare for these geopolitical risks, national capabilities are required to detect signs of nuclear proliferation around the Korean Peninsula. In particular, there is a need for constant monitoring of nuclear activities for peaceful purposes to ensure that those are not redirected for military purposes.

To assess the risk of nuclear proliferation in neighboring countries, technical and political evaluation factors were considered with the concept of timely warning in the Nuclear Non-Proliferation Act (NNPA) enacted by the US Congress in 1978. Technical evaluation factors were derived from keywords such as nuclear materials, nuclear facilities, and nuclear technology, whereas political evaluation factors were derived from keywords such as economic feasibility, relations with neighboring countries, and related treaties and agreements. Open-source data that is relatively easy to collect was used for deriving each evaluation factors and it was collected using real-time web-crawling techniques. To improve the reliability of collected data, a deep-learning techniques based expert model were applied. Finally, the risk of nuclear proliferation in neighboring countries is derived as ‘level’. Therefore, this study aims to 1) develop a model that can collect, process, and analyze open-source data related to nuclear activities by linking them with deep-learning models, and 2) develop an ‘integrated platform’ for assessing nuclear proliferation risk in neighboring countries. It is meaningful to process into useful data (intelligence) through expert model based on various types of structured and unstructured open data (information) and result of evaluation can be used as a basic tool to support decision-making in the process of establishing and implementing nuclear policies related to neighboring countries in the future.

Acknowledgements

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Keywords: Open-source, Non-proliferation, Web-crawling, Text mining

A Study on the Requirements and Design for Development of Decision Support System for Nuclear Export Control

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Strategic items utilized in the development of weapons of mass destruction (WMD) are strictly regulated under nuclear non-proliferation regimes, with experts in the nuclear industry responsible for their classification. As technology advances and international cooperation grows, the Korean government has implemented increasingly intricate export control policies to manage both tangible and intangible transfers of these strategic items, which can arise through various channels.

However, the surge in international technology exchanges has led to challenges, including delays in the classification and licensing processes, resulting in less efficient execution and completion of export licensing. Furthermore, while many export organizations possess extensive classification histories, they often underutilize this data due to undisclosed classification criteria.

Recognized government-certified organizations, such as those with Compliance Program (CP) certification, implement internal management systems. The government actively encourages businesses to obtain CP certification to autonomously manage strategic items in their international trading operations.

In response, this paper presents a system model and general requirements designed to automatically analyze documents pertinent to the classification of strategic items. By integrating disparate data points and harnessing historical information more effectively, the system aims to aid exporters in their decision-making processes. The paper covers the approach, data collection and analysis, as well as the derivation of general requirements.

The findings of this study will guide the development of a decision support system intended to streamline Korea Atomic Energy Research Institute (KAERI)'s export control procedures, enhance practical efficiency, and alleviate administrative and economic burdens. Ultimately, this endeavor is poised to reinforce both domestic and foreign technological competitiveness in the management of strategic materials.

Keywords: Export control, Trigger list, Strategic item, Classification, Non-proliferation

A Study on the Revision of the Regulation on the Export and Import Approval for Nuclear Materials to Improve the Nuclear Nonproliferation System

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The Regulation on the export and import approval for nuclear materials is established under Article 107 of the Nuclear Safety Act and Article 12 of the Foreign Trade Act, as notified by the Nuclear Safety and Security Commission (NSSC) based on an Integrated Public Announcement. This guideline mandates prior verification for importing and exporting nuclear materials, ensuring that only those with clearance or approval (exemption) can proceed. Despite its role in linking the obligations under the Nuclear Safety Act with the Integrated Public Announcement and confirming the duties of ‘nuclear material users’, the system’s enhancement is necessary from a nuclear non-proliferation regulatory perspective. This paper aims to review and propose improvements to the approval and exemption procedures for the import and export requirements of nuclear materials, including examination standards and required documents.

Firstly, the application and exemption processes are so similar that the administrative benefits of exemption are minimal, and the criteria for exemption eligibility are unclear, posing practical application challenges. Additionally, the lack of a specific provision for exemption application in the Public Announcement presents a limitation. Clear definitions for exemption and exclusion criteria and partial quoting of the Public Announcement in the application process could enhance efficiency.

Next, a revision is necessary to encompass all procedures and obligations that ‘nuclear material users’ must comply with before import and export, as defined in the Nuclear Safety Act. The current guidelines include peaceful use, compliance with IAEA international transfer reporting obligations, and safe use as criteria. However, there is no mechanism to verify the domestic management of small quantities of nuclear materials, leading to potential loopholes in managing these materials and complying with bilateral Nuclear Cooperation Agreement. Adding confirming standards to assess qualifications or readiness for the safeguards before import could prevent this.

Furthermore, a foundation for reviewing the implementation of agreements signed with the trading country and ROK is necessary. It requires detailing the origin information for each nuclear cycle in the import requirement verification documents and adding confirmation points for adherence to import and export procedures under agreements. Changes in approval criteria for specific nuclear materials under one effective kilogram, which are not subject to IAEA international transfer prior-reporting but frequently require prior notification with import and export approval verification, are also needed to reduce unnecessary administrative burdens.

Lastly, improvements to the application forms are essential. Simplifying the forms by removing irrelevant information and incorporating new confirmation information or allowing importers and exporters to mark certain information that is difficult for the verifying agency to judge could be considered.

In conclusion, to align with the regulatory objectives of nuclear non-proliferation and systematically implement the management of nuclear materials, a revision of the verification procedures, examination standards, and required documents is necessary.

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Keywords: Export and import control, Nuclear material, Nuclear non-proliferation

Proposal of the Nuclear Material Accounting Method for Molten Salt Reactors

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As Molten Salt Reactors (MSRs) gain traction as a promising next-generation nuclear energy technology, ensuring accurate and efficient measurement of nuclear materials, particularly plutonium, is imperative for nuclear safeguards.

Currently, ongoing research efforts are focused on Nuclear Material Accountancy (NMA) using radar techniques, specifically the LIST (Liquid Integral Scanner and Transmuter) method, alongside the application of Destructive Assay (DA) techniques through sample collection. While these methodologies have provided valuable insights, further investigation is warranted to refine and enhance NMA processes during MSR operation. However, there are some marked limitations in DA techniques, where it affects the materials physical integrity and taking samples is limited during the operation of an MSR which can act as a disadvantage in safeguards perspective. Therefore, Non-Destructive Assay (NDA) methodology is gaining strength as NDA will give a solution of a material in its complete form. Despite the uncertainty of NDA, following the necessity of a mobile and fast measurement of NMA, this research will focus on using NDA to measure Special Nuclear Materials (SNMs) such as plutonium and uranium regarding MSR.

Serpent 2 will be used to model our target MSR, Molten Salt Reactor Experiment (MSRE), a research reactor at the Oak Ridge National Laboratory (ORNL). Our approach will involve tracking the changes in graphite located at the outside of the core and analyzing the correlation between these graphite changes and the presence of SNMs. Specifically, by evaluating the isotopes that composite graphite, such as boron, through this approach we can retroactively trace the degree of the MSR depletion. Also, by analyzing the concentration change of SNMs as a function of burnup, we can quantify the amount of SNMs in terms of NMA.

Through this study, it is anticipated that inspectors will be able to indirectly evaluate burnup of the MSR, for example, by placing the MSR depletion indicator outside of the core. Ultimately, we seek to develop a liable method to secure NMA in MSR that will allow for discussions on the safeguards approaches, promoting peaceful utilization and enhancing nuclear non-proliferation.

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Keywords: Molten salt reactor (MSR), Nuclear material accountancy (NMA), Retroactive trace method, Safeguards approaches

Quality Management System for KINAC's R&D Processes Based on ISO9001:2015

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The importance of qualifying and sharing research outcomes to the next step advancement has been heightened and supported by laws such as the 'National R&D Innovation Act' and the 'Act on the Performance Evaluation and Management of National R&D Programs'. KINAC has, therefore, developed a new quality management system (QMS) based on ISO 9001:2015 for R&D outcomes. The system consists of a set of documented manual, processes, and instructions with the responsibility for ensuring that quality objectives are met with the organization. It involves defining quality policies, implementing quality control measures, and continuously monitoring and improving processes. For the quality management system in the field of R&D and outcomes management, the following steps were taken according to the procedures for QMS system establishment: conducting an initial assessment of the organizational structure and current status, defining the objectives, and determining the applicable scope of the QMS. Additionally, a QMS manual was drafted, and a document system, including processes and instructions, was developed for standardization. The efforts were focused on incorporating the existing institute's regulations and procedures into the QMS.

As a result, twenty-nine quality documents were prepared and categorized into four parts: general management, research planning, research and evaluation, and R&D outcomes management. The next step was implementing the developed QMS to the organization to evaluate the effectiveness and the efficiency of the system. A total of a hundred and four record templates were listed for the QMS. The 'records' refers to documented information that provides evidence of conducted activities or achieved results. The records are crucial for ensuring accountability, traceability, and continuous improvement of the organization's QMS. The internal audit and management review were also conducted to evaluate the effectiveness of QMS, identify areas for improvement, and ensure continuous compliance with ISO 9001 requirements. The implementation of quality management standards and procedures for R&D in KINAC could lead to enhanced research practices and increased efficiency in conducting R&D activities.

Keywords: Quality management system, Research and development, ISO 9001:2015, Quality manual, Processes, Records

Developing R&D Outcomes Management System for Effective Knowledge Administration of the Organization

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The ‘Act on the Performance Evaluation and Management of National R&D Programs’ emphasizes that research institutes should continue to manage the research outcomes even after the R&D program is finished. It is also recommended that research institutes establish a database of research outcomes and utilize it by linking with a national information system. KINAC has conducted various R&D projects related to nuclear non-proliferation and nuclear security. Managing research outcomes and ensuring their qualification have become more important since the scope and scale of KINAC’s R&D have gradually expanded and intensified.

In the present study, the ‘Integrated Research Outcomes Management System’ was developed to comprehensively manage institutional research outcomes. The system platform was designed to accommodate various types of research outcomes, including patents, copyrights, designs, standards, trademarks, utility models, software, book copyrights, and information on technology transfer.

Additionally, another function for R&D hardware was developed in order to continuously manage and control equipment along with its technical information came from the national R&D projects. The electronic lab notes were also developed to ensure the transparency of research processes, easily access R&D information, and effectively share R&D outcomes with researchers in KINAC.

Initially, enabling uploads of previous R&D document was prioritized to diversify the format of lab notes and allow continuous recording and accumulation of project-specific research processes. By providing the centralized platform for managing research outcomes, it is expected to increase efficiency in tracking, assessing and disseminating the research results, thus saving time and resources.

Keywords: Integrated research outcomes management system, IPRs, Equipment management, Lab notebook

Economic Feasibility Analysis for Nuclear Weapons Programs

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North Korea's nuclear tests and frequent provocations of force continue to shape public opinion in South Korea toward its nuclear arsenal. Surveys of South Koreans conducted by various research organizations have found relatively positive public opinion regarding nuclear armament. When asked why a nuclear arsenal is necessary, 39% of respondents said it was to "prepare for threats other than North Korea", 26% said it was to "enhance South Korea's international standing", and 23% said it was to "prepare for threats from North Korea". It can be understood as the illusion of nuclear weapons in political and diplomatic terms rather than the perceived threat of North Korea. The reason for these findings is that South Koreans do not recognize the benefits and harms of nuclear weapons, and to improve this; we will analyze the economic feasibility of nuclear weapons.

The economic effects of nuclear weapons development are manifested in so many different ways in so many different sectors that it is impossible to consider them all. Therefore, we analyzed the economic effects of military and energy security, which are easier to understand intuitively compared to other fields. Of course, there may be benefits and disadvantages that were not covered in this study, but above all, as a rogue state that deviates from international norms, such as North Korea, is expected to be isolated from the international community, it is hoped that this study will help foster a proper awareness of nuclear weapons and establishing a culture of nuclear nonproliferation in South Korea.

Keywords: Nuclear non-proliferation culture, Economic feasibility analysis, North Korea

DPRK's ELWR Analysis Based on VVER Light Water Reactor Fuel

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Since October of last year, drainage has been observed near DPRK's experimental light water reactor (ELWR). North Korea has historically used the Yongbyon 5 MWe reactor to produce plutonium, and the ELWR could also be operated to produce plutonium. This assessment assumed the nuclear fuel used in the Russian light water reactor VVER to account for possible technology transfers in North Korea's relationship with Russia. Considering the size of DPRK's ELWR, we evaluated the decay heat and plutonium production after discharge and cooling using nuclear fuel of the VVER-440 type. Plutonium production restriction is essential for nuclear nonproliferation, and decay heat is important for the timing of spent fuel handling.

SCALE ORIGIN was used for the depletion calculations, assuming ELWR thermal power with VVER-440 fuel. The plutonium yield per ton of heavy metal was assessed to be 2-3 times higher for the VVER reactor but lower in terms of quality (Pu-239/(Pu total isotopes)). The ELWR spent fuel was evaluated to require about three years after discharge to reduce it to the same level as the decay heat after 30 days of cooling the spent fuel discharged from a 5 MWe reactor.

The assessment in this study assumes idealized operation of the reactor so that a 5 MWe reactor would operate for three years and cooling time for 30 days, while an ELWR would operate for 120 days and cool down for three years, suggesting that ELWR could be used as an alternative to plutonium production. ELWRs need to be considered for plutonium production and as a means to other ends.

Keywords: ELWR, VVER, Decay heat, DPRK

Consideration of Artificial Intelligence Technology's Impact on Nuclear Non-Proliferation

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Recent advances in generative language models, such as ChatGPT, are revolutionizing AI in various fields, but there are also many concerns about the risks. In particular, as a technology that has the potential to facilitate the proliferation of sensitive technologies, its risks are being reviewed and controlled. The U.S. Department of state International Security Advisory Board (ISAB) analyzed the impact of rapidly developing AI technologies on key security issues.

However, advances in data science and computing power, such as AI technologies, also allow for the development of additional proliferation indicators that utilize new data sources. In other words, while traditional proliferation detection techniques have been limited by their focus on specialized nuclear materials and equipment used exclusively for nuclear weapons development, it is possible to overcome these limitations to detect covert nuclear activities. In preparation for on-site inspections (OSIs), the CTBTO is building a monitoring system using airborne and spaceborne multi-spectral imagery (MSIR) to classify inspection areas and detect changes. The system utilizes AI for more efficient analysis, applying pixel-based classification (unsupervised/supervised machine learning), object-based classification (decision rules and fuzzy logic), and multi-variate alteration detection (MAD). The IAEA is exploring the application of AI in two safeguards activities, spent fuel verification and remote monitoring, to improve the effectiveness of safeguards activities. We are looking to use gamma spectroscopy or Cherenkov imaging data for spent fuel verification. It is not mature enough to make autonomous judgments, but we continue developing it. The next example is remote surveillance with AI. Intelligent image analysis technologies such as AI can automatically track and monitor special events, simulate or build digital twins of similar facilities, and efforts are being made to develop the technology in various aspects.

Korea is also trying to continuously analyze signature data in neighboring countries' nuclear activities, and it is necessary to develop Korea's own AI-based big data platform to detect neighboring countries' nuclear activities. In particular, since nuclear nonproliferation requires complex analysis of technology, politics, diplomacy, and trade, AI technology is expected to identify subtle signs of strategic intentions to detect the transition to military purposes.

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Keywords: Nuclear non-proliferation, Artificial intelligence, Remote monitoring system

Current Status Review of AUKUS: Concerns and Responses for Nuclear Nonproliferation

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On September 15, 2021, the United States, United Kingdom, and Australia announced the launch of AUKUS, a trilateral security agreement in the Indo-Pacific region, named after the initials of the three countries. The parties to AUKUS expect the agreement to contribute to maintaining peace and stability in the Indo-Pacific region. Through AUKUS, the U.S. and UK will assist Australia in acquiring nuclear-propulsion submarines. However, the fact that the fuel for the U.S. and UK nuclear-propulsion submarines is highly enriched uranium has raised nuclear proliferation concerns in some countries. In particular, China has argued that the transfer of weapons-grade uranium from the U.S. and UK, as nuclear weapon states, to Australia, as a non-nuclear weapon state, violates the objectives of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). Responding to these concerns, the U.S., UK and Australia have cited their obligations to fulfill nuclear nonproliferation. They are also working to set a strong precedent for countries seeking to acquire nuclear-propulsion submarines in the future. While the AUKUS is not in clear violation of NPT, the possibility of non-compliance with safeguards and the risk of nuclear proliferation cannot be completely ruled out. As the international community has differing opinions on Australia's nuclear-propulsion submarines in the current situation, this paper will examine AUKUS from a nonproliferation perspective. This paper will first summarize the concerns of several countries, including China, Russia, and Iran, about nuclear-propulsion submarines. Second, it will examine similar cases where non-nuclear weapon states have sought to acquire nuclear-propulsion submarines. Third, this paper will summarize key implementation plans of AUKUS, focusing on how the three countries will prevent potential diversion of nuclear materials and the exploitation of loopholes in the Comprehensive Safeguards Agreement. In conclusion, a review of relevant cases and plans to introduce nuclear-propulsion submarines to Australia shows that the process will be conducted to minimize proliferation risks while strengthening the nuclear nonproliferation regime.

Keywords: AUKUS, Indo-pacific, Nuclear-propulsion submarine, Nonproliferation

Analysis of the Nuclear Material Accountancy System for Bulk Handling Facilities

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The Korea Atomic Energy Research Institute (KAERI) has reported 12 nuclear facilities as subject to IAEA safeguards. These safeguards facilities are categorized into Item Counting Facilities and Bulk Handling Facilities based on the characteristics of nuclear material accountancy for each type and form of facility. KAERI has established and operates a safeguards implementation system, to efficiently and effectively comply with international agreements such as the ROK-IAEA Safeguards Agreement and Additional Protocol, bilateral nuclear cooperation agreements, and domestic nuclear safety laws specified in nuclear material accountancy and safeguards duties. The nuclear control management department oversees the safeguards duties for KAERI's 12 nuclear facilities, which vary in type and nuclear material accountancy characteristics. Each nuclear facility appoints a nuclear material accountancy manager and personnel to not only implement the nuclear material accountancy tasks of the facility but also to comply with domestic and international safeguards obligations.

Among the basic obligations of nuclear facilities regarding safeguards, the most fundamental are the duties to manage nuclear materials accountancy and to report on safeguards information. To this end, KAERI has developed and operates the KAERI Safeguards Information Treatment System (KASIS). One of the most critical functions of KASIS is its near real-time nuclear material accountancy capability. In the past, when there were changes in the inventory of nuclear materials at a facility, the accountancy manager would provide a hard copy of the inventory change details to the nuclear control management department, which would then enter the data into a computer system. However, since the implementation of integrated safeguards in 2008, in response to IAEA's Random Interim Inspections (RII), the accountancy managers of each nuclear facility are now required to enter the records of inventory changes into KASIS on the same day they occur.

Item Counting Facilities and Bulk Handling Facilities apply different methods and procedures for material accountancy, so the methods and contents of the material accountancy reports submitted to the IAEA also differ. This paper examines the differences in material accountancy between these facilities, the current status of material accountancy at KAERI facilities, and potential improvements.

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Keywords: IAEA safeguards, Bulk handling facility, Nuclear material accountancy system

Development of Cybersecurity Training Equipment Based on Nuclear Power Plant Control and Instrumentation Systems

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In March, a significant cybersecurity breach unfolded at Duvel, a renowned brewery in Belgium, where a ransomware attack precipitated the complete shutdown of its production facilities. The attack was orchestrated by an individual or group known as “Stormus”, who infiltrated Duvel’s IT infrastructure, deployed ransomware, and consequently exfiltrated sensitive corporate data. This incident evokes memories of the infamous 2010 Stuxnet assault on the uranium enrichment facility in Natanz, Iran, thereby underscoring the paramount importance of cybersecurity and the profound implications cyberattacks hold for national infrastructure. Despite the ongoing emphasis on the criticality of safeguarding industrial control systems, the establishment of an infallible defense strategy remains elusive.

Amidst the escalating cybersecurity threats targeting computer systems, the safeguarding of industrial control and physical protection systems within nuclear facilities has emerged as a pressing concern. In response to the heightened demand for specialized cybersecurity training within the realm of nuclear facilities and the imperative to augment the proficiency of regulatory personnel, the Korea Institute of Nuclear Nonproliferation And Control (KINAC), has been diligently working to curate an array of educational programs tailored to these needs. This research initiative led to the development of an innovative training apparatus, leveraging digital simulations of the control and instrumentation systems characteristic of nuclear power plants. The primary objective was to delineate cyberattack trajectories, catalyze the implementation of security measures, and obstruct residual attack vectors. The design was meticulously crafted to confront five predominant cyberattack vectors: physical access, logical access, wireless access, the utilization of portable media and devices, and vulnerabilities inherent in the supply chain. Employing this sophisticated training equipment, the curriculum was designed to enhance trainees’ comprehension of diverse cyber threats and impart knowledge on both technical and administrative security strategies through meticulously crafted training scenarios.

The pedagogical tools and scenarios developed through this scholarly endeavor are anticipated to significantly contribute to the fortification of cybersecurity competencies within nuclear facilities. In light of the perpetual evolution of cyber threats, it is paramount to devise educational programs that not only stay abreast of these changes but also accurately mirror the dynamics of real-world settings. Consequently, this paper posits a strategic framework for the advancement of future cybersecurity educational programs, advocating for perpetual updates and the refinement of practical training modules to facilitate more effective and impactful learning outcomes.

Keywords: Cybersecurity, Cyberattack, Training equipment, Training scenarios

Analysis of the Status of Findings Issued in NMAC Inspections and Suggestions for Improvement Directions

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Since revising the Nuclear Safety and Security Commission (NSSC) notice in 2015, the system for Nuclear Material Accounting and Control (NMAC) inspection has been reorganized to examine operators' compliance with NMAC regulations. The inspection results in issuing findings for violations of laws, licensing conditions, technical standards, and procedures, and recommendations are issued for areas needing improvement or supplementation. Mainly, as follow-up actions by operators are mandatory for findings, providing clear evidence for the operators' understanding is crucial. This paper aims to analyze the current status of issued findings to derive directions for improvement. The issuance status of findings for six inspection sections - 'Organization and Roles', 'Record and Reporting', 'Measuring System', 'Accounting Control Procedure', 'Education and Training', and 'Others' - was analyzed. Additionally, the findings issued annually from 2015 to 2023 were examined. Analyzing issues identified through NMAC inspections is essential to identify and improve problems in implementing NMAC. Mainly, ongoing outreach activities are necessary to improve the 'Record and Reporting' and 'Accounting Control Procedure' sections, which comprise the majority of issued findings. Additionally, before requesting the issuance of findings, it is crucial to hold internal review meetings involving inspectors and managers to perform mutual verification and ensure regulatory consistency. Through such processes, it is necessary to enhance the quality of safeguards regulations implementation to prevent confusion among operators and implement transparent rules.

Keywords: Safeguards, NMAC, Regulations, National inspection

Components of Cybersecurity Incident Response Records

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According to Article 14 (Records and Storage) of the Act on Physical Protection and Radiological Emergency (hereinafter referred to as the APPRE), licensees must record matters related to the physical protection of nuclear facilities and retain them at each facility. Requirements related to physical protection are described in Article 9 (Records and Storage) and Annex Table 2 of the Enforcement Rules. Annex Table 2 consists of a total of 10 requirements. Among these, the tenth requirement is about a record of cyber security incident response to threats such as unauthorized removal of nuclear materials and nuclear facilities.

However, there is no specific description of what cybersecurity incident responses are recorded or how they are recorded. Furthermore, no cybersecurity incident to refer to have occurred so far in Korea. The only cybersecurity incident was a case where operational data had been leaked from some nuclear facilities. This is out of regulatory scope of the APPRE.

Nevertheless, it may be possible to infer the purpose of recording and retaining cybersecurity incident response, and suppose what and how content must be recorded. The first reason that could be inferred is the analysis of the incident itself. In other words, it is to analyze the cause and impact of the incident. The second reason is the analysis of the licensees' response to the incident. A regulatory body must want to reflect the lessons learned through these two types of analyses in regulatory positions, enactment and revision of laws and notices.

According to this perspective of an incident and response, it is clear that licensees must record each step of responses; detection, analysis, isolation, removal, and recovery. Specifically, who are involved, what decisions are made, why those decisions are made, and documents such as technical drawings and procedures referred to in the decision-making are the practical examples of records. Without these detailed records, licensee and a regulatory body cannot reenact the cyber security incident and response, and can get only fragmental information.

In this paper, we inferred the purpose of cybersecurity incident response records and described the detail of records. We hope that the accurate purpose of cybersecurity incident response record is described in the APPRE in near future.

Keywords: Cyber, Incident, Response, Records retention

Simulation Study to Apply a Compton Suppression Technique on CZT-based Spent Nuclear Fuel Characterization System

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A computational simulation study was conducted to design a Compton suppression-based gamma-ray detection system that can be used in a spent nuclear fuel inspection. The proposed CZT-based spent nuclear fuel characterization system has the advantage of being able to acquire high-resolution gamma-ray energy spectrum without cooling system. However, the count rate limit of the CZT detector is low, so an additional shielding system is needed when used for spent nuclear fuel inspection. The Compton suppression is a technique that reduces background gamma-ray by eliminating events caused by Compton scattering, and requires two detectors including a main detector and a guard detector, for anti-coincidence measurement.

We conducted a study to derive the type and geometry of a suitable guard detector to apply the Compton suppression technique on a CZT-based spent nuclear fuel characterization system. The system was configured to surround guard detector and lead shielding in that order outside the CZT detector. The MCNP 6.2 code was used to simulate the radiation transport and the Compton suppressed gamma-ray energy spectrum was obtained by simulating the anti-coincidence measurement. The peak to Compton ratio and reduction factor were derived through gamma-ray spectrum analysis to evaluate the performance of the Compton suppression system. A Compton suppression-based gamma-ray detection system suitable for the spent nuclear fuel inspection was designed by evaluating the performance of background suppression according to the change in the geometry of the guard detector and main detector. This system will be able to increase the accuracy and efficiency of analysis by reducing the influence of background gamma rays other than the gamma ray peak used in gamma-ray spectrum analysis.

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Keywords: Compton suppression, Spent nuclear fuel, Detector, Gamma-ray, Anti-coincidence

2분과

사용후핵연료 처분전관리 (ORAL)



Laser Induced Breakdown Spectroscopy (LIBS) and Its Success in Used Nuclear Fuel Disposal and Reprocessing Technology

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Imagine being able to analyze the composition of virtually any material, from solid rock to molten metal, without any need for extensive sample preparation. That's the incredible power of Laser-Induced Breakdown Spectroscopy (LIBS), a cutting-edge analytical technique that's been turning heads in the scientific community and beyond. LIBS is all about harnessing the awesome energy of a laser to create a miniature plasma explosion within the material being analyzed. This plasma emits light that reveals the elemental fingerprint of the sample. LIBS can do this with a high degree of accuracy and speed, making it a game-changer in fields like materials science, environmental analysis, and now, increasingly, nuclear technology. LIBS's versatility and non-destructive nature have caught the eye of researchers eager to tackle some of the biggest challenges in the nuclear field. Imagine being able to analyze the composition of used nuclear fuel without having to break it apart first – that alone could revolutionize how we handle and process this potentially dangerous material. Or consider the possibilities for improving the safety and efficiency of nuclear reactors by constantly monitoring their internal components for signs of wear or chemical changes.

Of course, applying LIBS to the nuclear realm isn't without its challenges. These ultra-powerful lasers and their ability to create stable plasmas within nuclear fuels are still areas of active research. But the potential rewards – in terms of both technological advancement and public safety – are immense. That's why it's exciting to delve into the world of LIBS and explore its journey into the nuclear field, especially used nuclear fuel disposal and reprocessing systems. We'll discuss the cutting-edge research being done, the successes achieved so far, and the hurdles still to overcome. Maybe we'll uncover the next breakthrough that takes this amazing technology from the lab to the reactor.

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Keywords: LIBS, Nuclear fuel cycle, Used nuclear fuel disposal, Reprocessing technology

Surface Peening Technologies to Mitigate Stress Corrosion Cracking in Dry Cask Storage Stainless Steel Canister for Spent Nuclear Fuel

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The safe, secure, and reliable storage of spent nuclear fuel is essential to ensure sustainability of the current light water reactor (LWR) fleet worldwide. In the absence of permanent geologic repositories in the United States, as in many nations, there are inevitable demands for interim storage of spent nuclear fuel in dry cask storage system (DCSS) for extended periods of time. The major component of DCSS is an austenitic stainless steel (304 or 316 grade) canister providing physical separation of the radioactive materials from external environment. However, heat-affected zone (HAZ) around the fusion-welded regions of the canisters are susceptible to pitting corrosion, eventually resulting in catastrophic chloride-induced stress corrosion cracking (CISCC). This phenomenon is considered as one of primary life-limiting factors in the long-term operation. The concern still exists given the potential long-term operation of the interim storage although obvious CISCC is reported in the current DCSS.

This presentation will provide a brief overview of state-of-the-art surface peening technologies to address the challenges of CISCC in DCSS canisters. The approaches are pulsed water jet peening (PWJP), ultrasonic nanocrystalline surface modification (UNSM), and laser shock peening (LSP). For each surface peening technology, the most dominant process parameter was investigated, namely water jet traverse speed for PWJP, laser power density for LSP, and applied static load for UNSM. The substrate material was chosen as sensitized 304H stainless steel to simulate the condition of the canister. Sensitization was performed at 650°C for 48 hours to demonstrate the microstructure of the HAZ of the welds. The high carbon 'H' grade was selected to emphasize the material sensitization effects. Compositional and microstructural information of the samples was examined using scanning electron microscopy (SEM) in conjunction with energy dispersive X-ray spectroscopy (EDS). X-ray diffraction (XRD) was used to identify the phase and to measure residual stress via $\sin^2\Psi$ method. Corrosion potential and pitting potential of the surface peened samples were measured using electrochemical anodic polarization testing while the samples submerged in 3.5wt% NaCl solution. X-ray photoelectron spectroscopy (XPS) was also utilized to evaluate effects of ablative layer on surface chemistry evolution of the LSP samples.

Keywords: Used nuclear fuel, Dry cask storage system, Surface peening technologies, Chloride-induced stress corrosion cracking

Experimental Evaluation of Performance for Gamma Emission Tomography to Inspect Partial-Defect Within Pressurized Water Reactor-Type Spent Nuclear Fuel

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As the amount of electricity produced by a nuclear power generation increases, management and monitoring of the Spent Nuclear Fuel (SNF) is also emerging. IAEA has developed the varying method to inspect a SNF, however, the most methods could not detect partial-defect within SNF at pin-by-pin level. Previously, we optimally designed the Gamma Emission Tomography (GET) instrument, named Yonsei Single-photon Emission Computed Tomography version 2 (YSECT.v.2), to detect partial-defect through cross-section imaging, and its feasibility was evaluated based on the Monte Carlo methods. The objective of the current study is to fabricate the prototype YSECT.v.2 instrument and to experimentally validate its performance.

Based on the optimization study, total of four GAGG-based detection modules were fabricated. The data acquisition module was composed of ASIC-based data processing circuit from the PETsys Electronics Co. Ltd. Additionally, waterproof housing and Peltier device-based heat reduction module were employed for the stable operation. To evaluate performance of the YSECT.v.2 instrument, projection and tomographic images were obtained using the mock-ups of fresh nuclear fuel and ¹³⁷Cs check sources. The activity of each source was calculated to be 0.01 and 5 μ Ci, respectively, and dominant energy was 185 and 661.7 keV. Filtered Back-Projection (FBP) algorithm was employed to reconstruct tomographic image. Moreover, the thresholding approach, which was developed in the previous study, was applied to improve tomographic image quality.

Based on the results of projection image for mock-ups of fresh nuclear fuel, the Signal-to-Noise Ratio for each detection module was calculated to be 2.50, 3.82, 3.31, and 2.89, respectively, and Full-Width at Half-Maximum was 10.8 mm. With the FBP algorithm and thresholding approach, the source distribution was clearly distinguished in spite of the low-activity of mock-ups of fresh nuclear fuel. As similar experiment condition, the tomographic images for ¹³⁷Cs check source were obtained. Given that the tomographic images were reconstructed using only 16 projection images, the aliasing artifact was incurred. Nevertheless, the pattern classification accuracy was calculated to be about 100% without thresholding approach.

In the current study, we fabricated a GET instrument for detection of partial-defect in SNF, and further, we experimentally evaluated its performance using mock-ups of fresh nuclear fuel and ¹³⁷Cs check sources. Based on the results, we believe that the fabricated instrument could be effectively employed to inspect and detect possible replacement scenarios within SNF. In the future, performance of the YSECT.v.2 instrument will be experimentally validated using the mock-ups of SNF in water.

Acknowledgements

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Keywords: Safeguards, Spent nuclear fuel, Partial-defect, Gamma emission tomography, Tomographic imaging

Al Cathode Based Actinides Separation Over Lanthanides Via Molten Salt Electrolysis

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Dry reprocessing is generally a process of separating actinides from fission product elements in a non-aqueous medium. Dry reprocessing is considered to be an essential technology for dealing with future fast reactor spent fuels, and molten salt system is the most commonly used medium. Molten salt based electro-refining technology proposed by American scholars has been promoted and followed up in Japan, South Korea, India and other countries. The molten salt electrorefining process can treat metal fuel and oxide fuel. However, the deposition potentials of Pu, minor actinides (MA) and rare earth elements (RE) on the liquid cadmium cathode (LCC) are close, so that the separation factors of Pu, MA and RE elements on the LCC are not high. In recent years, Institute of High Energy Physics of the Chinese Academy of Sciences has systematically carried out research on the separation of actinides (An) over lanthanides (Ln) by co-reduction to form aluminum alloys, and expects to apply this method to the dry reprocessing of spent fuels. The results show that Th, U and RE elements can be co-reduced with Al to form intermetallic compounds, and the application of active aluminum cathode can greatly improve the separation effect of An and Ln in the molten salt electrolysis system. However, based on the separation of An-Ln from solid Al cathodes, the cathode product is a solid Al-An alloy, which cannot be directly reused in fuel manufacture, and further separation of Al and An is required. In response to this challenge, our team developed an improved electrolytic refining process - "in-situ anode precipitation" technology. This separation process combines electrochemical oxidation and precipitation technology, no longer depends on the difference in deposition potential of the target element, but based on its solubility in NaAlCl₄ low-temperature molten salt. Experiments show that this technology can be successfully used in the separation of Al-U alloys. U is recovered at the anode in the form of UCl₃, and the recovery rate is close to 95%.

Keywords: Molten salt, Al cathode, Actinide, Electrolysis, Lanthanide

Long-Term Oxidation Behavior of Borated Stainless Steel (Neutron Absorber Based on Stainless Steel) in Spent Fuel Pool Environment

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High density storage rack in spent nuclear fuel pool is used frequently due to large amount of spent nuclear fuels which should be cooled before stored in dry spent nuclear storage. In this regard, maintain subcriticality of spent nuclear fuels was raised as an issue and borated stainless steel (BSS) was selected as structural material and neutron absorber for high density storage rack. The boron contained in BSS exists as secondary phases such as $(\text{Fe, Cr})_2\text{B}$ due to its low solubility in austenitic stainless steel. Furthermore, there are several limitations when it comes to demonstrating the long-term corrosion resistance in real-world environments, as it would require experiments spanning several decades. Therefore, accelerated corrosion experiments were conducted to gain insights into the long-term corrosion and oxidation behavior of BSS. These accelerated corrosion experiments were carried out under simulated SFP (spent fuel pool) conditions for simulation and prediction purposes. The accelerated corrosion experiments were conducted according to the Arrhenius equation, with acceleration factors and temperatures determined for the experiment.

The results from 6 months of exposure showed that the oxidation layer of the secondary phase was thicker compared to that of the substrate. Additionally, the secondary phase exposed on the surface oxidized in a porous structure, and Cr dissolution was observed. This is because, from a thermodynamic perspective, the oxidation of the secondary phase, Cr boride, is a stable reaction. Oxide layers of Ni were also formed in the secondary phase that did not contain Ni, indicating the possibility that the oxidized and porous secondary phase may have acted as a diffusion path for Ni.

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Keywords: Neutron absorber, Borated stainless steel, Accelerated corrosion experiment, Chromium boride corrosion, Cr dissolution

A Study on the Degradation Mechanism of Spent Fuels During Loss-of-Coolant-Accident Condition in a Spent Fuel Pool

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The paradigm of nuclear plant operations has changed to ‘safety is first’ since the Fukushima nuclear power plant accident in 2011, and the study on nuclear fuel is required when an accident occurs in the spent fuel pool (SFP). In this paper, the degradation mechanisms such as the air oxidation behavior and reaction rate of UO_2 pellets and Zircaloy-4 cladding were studied under of severe accident conditions in the SFP. This study aimed to predict the behavior of the air oxidation reaction of UO_2 and Zircaloy-4 cladding, assuming that under the severe accident caused by the loss of cooling function and/or coolant accident occurred in the SFP. In this study, A TGA was utilized, which is capable of measuring real-time reaction of UO_2 pellets and Zircaloy-4 cladding with air. The test temperature was set from 600°C to 1,400°C, and the weight change of the fuel specimen was measured in real time at each temperature.

The main results are as follows:

First, UO_2 tends to increase the rate of oxidation again as the temperature increases in the high temperature region, and unlike the low temperature region, the initial air oxidation reaction occurs rapidly up to about 2wt%.

Second, the weight increase behavior with oxidation reaction time with air initially showed a decrease in reaction rate with accident progression time, and followed the parabolic kinetic law, where the oxidation rate exponent was approximately 2.

Keywords: Spent fuel pool, Zircaloy-4 spent fuel cladding, UO_2 pellets, TGA, Loss-of-coolant-accident

Preliminary Thermal Analysis for the Design of an Experimental Device Aimed at Demonstrating the CANDU Spent Fuel Storage Silo

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CANDU nuclear reactors' spent fuel undergoes a storage process consisting of 6 years of wet storage followed by 50 years of dry storage before its final disposal. Silos are utilized as dry storage facilities during this phase. In Korea, silos have been in operation since 1987, and many are approaching the end of their licensing period.

The primary role of silos is to maintain the integrity of the stored nuclear fuel. Temperature is one of the key factors used to assess this integrity. The temperature of the nuclear fuel is related to oxidation, corrosion of components, and is a critical indicator for evaluating the mechanical properties of the materials. Therefore, assessing the temperature within silo facilities is crucial for demonstrating safety.

To date, computational analysis using computer codes has predominantly been used for temperature evaluation. Our research team is planning experimental validations to assess how closely these computational analysis models align with real. This paper introduces preliminary analysis results intended for designing these validation experiments. This analysis was performed using FLUENT, a commercial CFD code. Analyses were conducted on three inputs: single fuel bundle analysis, storage basket analysis, and silo analysis. The inputs for these analyses were derived from the realistic thermal analysis model developed in previous studies. Based on the preliminary analysis results, we have identified measurement targets and locations for the experiment.

Keywords: CANDU spent fuel, Dry storage, CFD calculation

Parameter Study of Radionuclide Release Rates for a Spent Fuel Transportation Cask Submerged in Deep Sea

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The current stage of spent nuclear fuel (SNF) management in South Korea is in the temporary storage phase, with preparations underway for the construction and system development of an interim storage facility. Considering the environmental and infrastructural situation within South Korea, it is highly likely that the interim storage facility will also be located along the coast. For these reasons, maritime transportation of SNF is deemed highly plausible. However, there currently exists no specific regulation in Korea for the maritime transportation risks assessment of SNF, necessitating the development of technologies to evaluate the impact of accidents occurring during maritime transportation.

Research is ongoing, utilizing computational fluid dynamics techniques to quantitatively assess the release rate of radionuclides from a submerged transportation cask lost at sea. The release rate of radionuclides must be calculated based on the flow passing through the damaged containment boundary. The type B transportation cask is a robust engineering system, and the breach generated in the damaged containment boundary of a cask even during a severe accident is expected to be very small. The simulation of the flow through this breach is challenging due to the significant scale difference between the deep-sea environment and the breach. To address this issue, a “Full Field Model” was developed to evaluate the flow inside and around a cask without breach, and a “Local Field Model” was developed to assess the flow through the breach generated in the damaged containment boundary of a cask. Sub-modeling techniques were employed to integrate the flow fields of the two models, using velocity and convective heat transfer coefficient values to combine the two models.

A parameter study was conducted using the developed model to evaluate the effect of various parameters on the release rate of radionuclides. This study included parameters related to the deep-sea environment, such as flow velocity and seawater temperature, as well as parameters specific to the SNF, like decay heat, and those related to the severity of the accident, such as the width of the breach generated at the containment boundary of cask. The research is aimed to identify critical parameters influencing the release rate of radionuclides, to facilitate the determination of conservative parameter settings. Notably, it was observed that the flow of seawater around the transportation cask significantly impacts the release rate. These results were compared with those obtained from another code developed by CRIEPI that only considers the natural convection within the transportation cask for benchmarking purposes.

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Keywords: Maritime transportation, Computational fluid dynamics, Risk assessment, Release rate, Spent nuclear fuel

2분과

사용후핵연료 처분전관리 (POSTER)



International Seismic Shake Table Tests for SNF Dry Storage Casks Update

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A series of full-scale seismic shake table tests of the spent nuclear fuel (SNF) dry storage systems have been planned by DOE spent fuel waste disposition (SFWD) program to determine the seismic loads acting on the SNF in dry storage systems in US. This testing work is led by the US Department of Energy's (DOE) Office of Spent Fuel and High-Level Waste Disposition, with testing being conducted for DOE by Sandia National Laboratories (SNL) in the US. KNF and CRI of KHNP are participating as international collaboration partner in this test program. For the tests KNF has provided the KNF 17ACE7 and PLUS7 surrogate test assemblies to SNL. The test will be conducted at the LHPOST6 shake table test facility operated by University of California at San Diego (UCSD) during the period of May to Aug. 2024. The test units consist of a NUHOMS 32 PTH2 canister, a mockup of a generic vertical cask, a mockup of a generic horizontal storage module, 4 surrogate fuel assemblies, and 28 dummy assemblies.

The seismic inputs for the tests will consist of ground motions (acceleration time histories) representative of Central and Eastern US, Western US and Korean sites. KHNP produced the ground motion considering the ground conditions in domestic PHWR dry storage facility and provided it to SNL.

Therefore, it is expected that the seismic loads acting on the SNF in dry storage cask can be identified through the tests on various spectra of the domestic seismic load as well as the US seismic loads, and based on these results, it can be used to design a domestic dry storage casks and evaluate the integrity of SNF during dry storage.

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

Preparation and Characterization of Metakaolin-Based Sorbent Beads for Iodine Capture

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In order to prepare a bead-type iodine adsorbent with high packing density and Ag^0 content, Na^+ beads were synthesized using two kinds of metakaolin with different Al/Si ratios and silicone oil as a spheroidizing medium. Ag^+ beads were prepared by impregnating the cured Na^+ beads into an AgNO_3 aqueous solution and then Ag/Na ion exchange. Na^+ beads were synthesized in various molar ratios of $\text{Al}_2\text{O}_3/\text{SiO}_2$, $\text{Na}_2\text{O}/\text{Al}_2\text{O}_3$, and $\text{H}_2\text{O}/\text{Na}_2\text{O}$ to increase the Na^+ content. Alkali activators, $\text{NaOH}/\text{Na}_2\text{SiO}_3$ and NaOH aqueous solutions were used for MASS (metakaolin activated by sodium hydroxide and sodium silicate solution) and MAS (metakaolin activated by sodium hydroxide) tests, respectively.

The properties of Na^+ and Ag^0 beads were analyzed through instrumental analysis such as EDS, XRD, FTIR, surface area and pore size analysis, crushing strength, and measurement of packing density and total iodine-capture capacity.

Ag^0 beads exhibited silver content of 41.7-45.5wt%, packing density of 0.57-0.77 g/cm³, and crushing strength of 1.5-8.8 N. In textural properties, Ag^0 beads had the BET surface area of 4.2 m²/g, the pore volume of 0.008-0.037 cm³/g, and the average pore size of 11.8-13 nm (desorption basis). The total iodine-capture capacities of Ag^0 beads were very high at 0.49-0.54 g-I/g-sorbent.

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Keywords: Metakaolin, Bead preparation, Sorbent, Silver nanoparticle, Iodine capture

Adsorption Characteristics of Se-79 From Spent Nuclear Fuel on CaO Pellets

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During the processing of spent nuclear fuel, various fission products are generated, including radioactive selenium, which has a lengthy half-life of 3.27×10^5 years and exhibits high mobility in aqueous solutions. Strict management measures are necessary to prevent its release into the environment. Selenium undergoes volatilization to selenium oxide when subjected to heat treatment in the presence of oxygen, necessitating research into the development of effective adsorbents for its capture. While previous studies have utilized calcium oxide to capture minute amounts of selenium oxide emitted from a fuel combustion, this study represents the first attempt to target on the capture of radioactive selenium volatilized from spent nuclear fuel. Calcium oxide pellets were prepared for this purpose, and their efficacy in capturing gaseous selenium was examined under various operational conditions. Furthermore, the safety of the adsorbent was assessed by evaluating its thermal and mechanical strength.

Keywords: Spent nuclear fuel (SNF), CaO pellet, Se-79, Adsorption

Reactive Distillation Process for the Strontium in Spent Nuclear Fuels Separated Via MgCl_2

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The separation of high heat-generating isotopes like Strontium-90 (^{90}Sr) and Cesium-137 (^{137}Cs) from spent nuclear fuels (SNFs) is a significant area of research in nuclear waste management. Especially, ^{90}Sr undergoes decay through Yttrium-90 to Zirconium-90 and has a relatively short half-lives. Moreover it emits substantial amounts of decay heat, making them prime candidates for removal to reduce the heat load and volume of waste requiring long-term storage.

To remove ^{90}Sr from SNFs, molten salt was utilized in KAERI. The Sr dissolves in the molten salt in the form of SrCl_2 and/or Sr_4OCl_6 . Among various molten salt, MgCl_2 exhibited not only the highest efficiency in separating of Sr but also possesses inert properties unlike KCl, which reacts with SrCl_2 to form KSr_2Cl_5 . Afterwards, it is essential to extract Sr in the form of oxide from the salt to produce immobilized forms suitable for disposal. This can be accomplished through reactive distillation using K_2CO_3 .

In this study, reactive distillation process for the Sr separated via MgCl_2 was employed. Due to the low concentration of Sr in the SNFs (0.121wt%), directly adding K_2CO_3 to the molten salt could result in significant possibilities of side reactions occurring. Therefore, the reactive distillation process was divided into two steps: the 1st step involves evaporating the all MgCl_2 and the 2nd step includes adding K_2CO_3 to make oxides through solid-solid reaction.

In the 1st step, the mixture of MgCl_2 and SrCl_2 was heated at 850°C for 2 h to evaporate the MgCl_2 under a vacuum (< 0.02 torr). From the XRD analysis showing only SrCl_2 after the 1st step, it appears that the distillation of MgCl_2 proceeded effectively. Additionally, ICP-OES analysis confirmed the presence of approximately 0.94wt% of residual Mg. The Mg was found to exist in the form of MgO and $\text{Mg}(\text{OH})_2$. After the 1st step, various amounts of K_2CO_3 (mole ratio of $\text{SrCl}_2:\text{K}_2\text{CO}_3 = 1:1, 1:2, 1:3$) were added for the reactive distillation to proceed. The mixture was heated at 550°C under 0.9 bar for 1.5 h for the reaction to occur. Subsequently, it was heated at 850°C under vacuum (< 0.02 torr) for 2 h to distill the reaction product KCl and CO_2 . As a result, in the 1:3 molar ratio, excess K_2CO_3 resulted in the formation of $\text{K}_2\text{Sr}(\text{CO}_3)_2$, while in the 1:1 molar ratio, insufficient K_2CO_3 led to the presence of unreacted SrCl_2 and the formation of Sr_4OCl_6 . The formation of SrO was confirmed in the 1:2 molar ratio, indicating that this ratio was the most suitable.

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

Formation of K_2ZrCl_6 in LiCl-KCl Molten Salt

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The chlorination process using $ZrCl_4$ in high-temperature molten salt has been studied as a method for chlorinating spent nuclear fuel. However, $ZrCl_4$ has a sublimation property that vaporizes at a temperature similar to the melting temperature of molten salt and solubility of $ZrCl_4$ in molten salt is very small. Therefore it is difficult to dissolve a large amount of $ZrCl_4$ in molten salt. However, it is known that once $ZrCl_4$ is dissolved in molten salt, it does not vaporize even at high temperatures and remains in the molten salt. $ZrCl_4$ above the solubility remains in the molten salt in the form of M_2ZrCl_6 . Here, M represents an alkali element. Therefore, in this study, a flange-type sealed reactor was fabricated to dissolve a large amount of $ZrCl_4$ in LiCl-KCl salt, to observe the reaction characteristics of LiCl-KCl salt and $ZrCl_4$. The salt used in the experiment was a LiCl-KCl eutectic salt, and for reference, an experiment using only KCl was also performed. Reaction characteristics were observed according to the molar ratio of $ZrCl_4$ added to the number of moles of KCl in LiCl-KCl salt. The molar ratios of $ZrCl_4$ to KCl were 0.5, 1, 2, and 3, respectively. The temperature at the reactor surface was above 500°C maintained at that temperature for 48 hours. To observe the effect of reaction temperature and reaction time, experiments were also conducted at a reaction temperature of 300°C and reaction time of 12 hours and 24 hours. After completion of reaction, the produced salt was recovered, and the characteristics of the reaction product were identified through XRD and ICP analysis. As a result of all experiments, K_2ZrCl_6 was identified in the produced salt, and a high concentration of Zr in the salt was observed. Therefore, from this study, it was found that high concentration of $ZrCl_4$ could be dissolved in LiCl-KCl salt by using a sealed reactor and controlling the reaction temperature, time, and number of reaction moles of $ZrCl_4$ added.

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

Evaluation of Alumina Scale Formed on Haynes 214 Alloy for Gas Corrosion Mitigation

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This study delves into the development of advanced heat treatment techniques tailored to bolster the stability and durability of the Haynes 214 alloy, an essential material used in electrolytic reduction processes as anode components. Pyroprocessing stands as a cutting-edge technology designed to extract uranium and transuranic elements (TRU) from spent nuclear fuel within high-temperature molten salt environments. Given the rigorous demands of pyroprocessing, materials employed must exhibit exceptional stability, particularly during the electrolytic reduction phase. Exposure to chlorine and oxygen gases at elevated voltages presents significant challenges, including corrosion and thermal shock, underscoring the critical need for stable materials in such environments.

The primary focus of this research lies in developing a precise heat treatment methodology, executed at 1,100°C, to facilitate the formation of a stable and corrosion-resistant Alumina scale on the surface of the Haynes 214 alloy during reduction reactions. Through meticulous SEM/EDS and FIB/TEM analysis, the thickness and integrity of the coating layer were meticulously evaluated, revealing notable enhancements in the alloy's durability and resistance to environmental stresses.

Furthermore, comprehensive corrosion testing was conducted under controlled exposure to chlorine and oxygen gases. The experimental outcomes elucidated that the Haynes 214 alloy, post heat treatment, exhibited consistent and stable operation, with the Alumina scale effectively safeguarding surface integrity against adverse environmental conditions. These findings not only validate the suitability of the Haynes 214 alloy for high-temperature applications but also underscore its potential for broader industrial uses where stability and durability are paramount.

Keywords: Haynes 214 alloy, Heat treatment, Electrolytic reduction, Alumina scale, Corrosion resistance

Comprehensive Studies on Characteristics of Spent Fuel From Small Modular Reactors Worldwide

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Small Modular Reactors (SMRs) are gaining attention for their role as a low-carbon energy source in climate change mitigation plans. These newer generation reactors are designed to produce electric power up to 300 MW(e) and are also utilized for non-electrical industrial purposes such as water desalination and industrial heat generation. According to the IAEA's booklet on Advances in SMR Technology Developments, published in 2022, there are over 80 SMR concepts currently in development, covering a wide range of reactor technologies.

While much focus has been given to aspects of SMR deployment such as reactor concepts, engineering, economics, infrastructure, safety, etc., the fuel cycle, and in particular the management of Spent Fuel (SF), appears to have had limited consideration. There is a need for increased focus on the back-end nuclear fuel cycle, storage, transportation, reprocessing, recycling, and disposal of spent nuclear fuel produced by SMRs.

SMR concepts encompass various designs, including evolutionary versions of Light Water Reactors (LWR-SMRs), High Temperature Gas Cooled Reactors (HTGRs), Liquid Metal Fast Reactors (LMFRs), Sodium Cooled Fast Reactor (SFRs) and Molten Salt Reactors (MSRs). The characteristics of the radioactive waste generated in SMR reactors vary depending on factors such as fuel type, fuel enrichment, burnup, refueling cycle, reactor size, selection of coolant/moderator, and whether the nuclear fuel cycle is open or closed.

Stanford University conducted an analysis of SMRs from Toshiba, NuScale, and Terrestrial Energy. It was anticipated that smaller reactor sizes would result in increased neutron leakage compared to conventional large reactors, leading to higher neutron activation and potentially higher radioactivity of SF.

Additionally, the Department of Energy (DOE) analyzed the characteristics from Light Water Reactor SMRs (NuScale Power's VOYGR™), Sodium-cooled Fast Reactors (TerraPower's Natrium™), and High-Temperature Gas-cooled Reactors (X-energy's Xe-100) in the backend nuclear fuel cycle. VOYGR™ is expected to increase the mass, volume, radioactivity, decay heat, and radiotoxicity of SF, while Natrium™ is anticipated to increase DU mass and radiotoxicity of SF while decreasing volume and mass of SF. Xe-100 is expected to decrease DU mass and radiotoxicity of SF, but increase volume of SF.

Previous studies have shown the diverse characteristics for different types of SMRs. Therefore, this study compared and analyzed radioactive waste, mass, volume, radioactivity, decay heat, and radiotoxicity of SF relevant to backend of nuclear fuel cycle management for various types of developed SMRs worldwide.

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Keywords: Small modular reactor, Backend of nuclear fuel cycle, Radioactive waste, Spent fuel, Depleted uranium, Radiotoxicity

Applying the Slip Casting for Uranium Oxide Block Fabrication

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The storage of SF (Spent Fuel) is critical issue for utilities not only a country but also worldwide. Nonetheless the national plan for SF management, when securing a disposal site and intermediate storage facilities is delayed, the economic costs and environmental burdens will increase for securing additional storage space for SF. Solidification technology for manufacturing Uranium oxide blocks is needed to safely disposal of discharged SF for research use.

In this study, we investigated the feasibility of applying the slip casting process used to manufacture pottery to the production of spent fuel oxide blocks. Before testing with radioactive materials, oxide blocks were successfully fabricated with CeO_2 powder, which has similar chemical properties to UO_2 . Based on this results, we fabricated a block with DU (Depleted Uranium) powder which is a simulated oxide SF. We tried to find the optimal conditions by dividing the process into six steps: preparing a binder, sludge preparation, injection/molding, demolding, wet drying, and sintering. Although there were slight differences from the optimal conditions of CeO_2 powder, we were able to find the optimal conditions for uranium powder through numerous trials and errors. In addition, a lot of trials and errors were made and solutions were proposed.

As a result, we succeeded in manufacturing Uranium oxide blocks with diameters of 30, 60, and 95 mm. If the quality is proven through mechanical tests of density, fracture strength, and leaching tests in the future, it can be established as a good spent nuclear fuel management technology.

Acknowledgements

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Keywords: Solidification, Slip casting, Uranium oxide block

Design and Performance Test of Vertical Sintering Furnaces

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Securing a disposal site and storage facilities for high-level waste such as SF (Spent nuclear Fuel) lead to the economic costs and environmental burdens. To reduce the environmental burden of spent fuel disposal, nuclide management technologies are needed that can selectively separate, capture, and dispose of major nuclides (long half-life, high heat release, and high mobility) contained in SF to improve disposal efficiency. After separating the major nuclides from the SF, only the powdered form of U_3O_8 remains. The final byproduct, U_3O_8 , must also be safely managed as a high-level waste because there is no complete separation of the major nuclides from the spent fuel.

In this study, a vertical type furnace was designed to manufacture Uranium oxide blocks. The sintering furnace was divided into seven design requirements: sintering object, sintering temperature, heating element, heating proof material, tube, thermocouple, and vacuum pump. The objects to be sintered are large oxide blocks which have about 20 to 70 times the weight of nuclear fuel pellets used in commercial reactors. Therefore, a horizontal sintering furnace would have bent and damaged the tubes due to the self-weight of the blocks at a high-temperature environment, so a vertical type furnace was designed. Due to the high operating temperature we used a heating element made of $MoSi_2$ called Kanthal Super which has high oxidation resistance. The sintering furnace manufactured according to these design requirements was verified through tests such as heating/cooling control tests and vacuum performance tests.

Acknowledgements

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Keywords: Uranium oxide block, Sintering, Vertical type furnace, Heating/cooling

Self-Calibration of the Differential Scanning Calorimeter

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

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

In this study self-calibration process of differential scanning calorimeter (DSC) was established to ensure the B-type uncertainty.

The self-calibration was performed using the standard reference material, SRM 720 (Synthetic sapphire, α -Al₂O₃, NIST) and correction factor was obtained. The correction factor was defined as the ratio of the specific heat value of the SRM reported in the certificate and the specific heat value measured using DSC. The uncertainty of the correction factor was also calculated. The correction factor was from 1-0.118 to 1+0.007 and uncertainty of correction factor was from 0.407% at 50°C to 0.401% at 390°C.

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

Acknowledgements

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Keywords: Differential scanning calorimeter (DSC), Standard reference material (SRM), Standard reference data (SRD), Uncertainty, Self-calibration, Nuclear fuel and materials data center

Comparison of Base Metal & Weld Metal for Canister Chloride-Induced Stress Corrosion Cracking Using DCPD & EBSD Measurement

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

In this study, a simulation test using the DCPD measurement method was performed based on the CISCC simulated accelerated test device built in the last step, and test results analysis and microstructure analysis according to each test condition were performed. As a result of performing a CISCC simulation acceleration test for about 400 hours at a test temperature of 50°C, 7% MgCl₂, and a load range of 1.00~1.75YS, it was confirmed that the voltage value in DCPD did not change for both the welding material and the base material, showing that cracks caused by CISCC were confirmed. It was confirmed that no tremor phenomenon occurred. It was confirmed that there was no change in crack length before and after the actual test, and in the case of 316 L material, it has been reported that the CISCC phenomenon does not occur, and the same result was confirmed in this study. Additionally, the crack area of the specimen after the test was analyzed with EBSD, and low-frequency fatigue crack propagation progressed to intragranular fracture in both the base material and the weld zone.

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

Hot Cell Examination for Measuring Rod Internal Pressure of Pressurized Dummy Fuel Rod

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Korea Atomic Energy Research Institute (KAERI) operates the Post Irradiation Examination Facility (PIEF), where various PIEs are conducted on PWR fuels to evaluate their properties, especially those observed after neutron irradiation. One of the key PIEs performed at the PIEF is the measurement of rod internal pressure. This test is essential for evaluating characteristics related to fission, and the resulting data is crucial for estimating hydride reorientation behavior in terms of spent fuel storage and management. Since the test is conducted on full-length spent fuel rods, the testing equipment is installed in a shielded hot cell, making equipment maintenance somewhat tricky and restricted. Additionally, this test can be performed only once for each fuel rod, and proficiency is required for the workers involved. In the present study, we conducted an examination using a pressurized dummy fuel rod. The purpose of this test was to inspect the equipment's condition, identify improvement opportunities for both the equipment and the testing procedures, and provide training for worker proficiency.

Keywords: Hot cell, Rod internal pressure, Fission gas, Post irradiation examination, Fuel cladding

Separation of Strontium Chloride From Uranium Oxide by Vacuum Distillation

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Highly toxic radio-isotopes and high heat emitting isotopes in spent nuclear fuel are the burden materials in the deep geological repository. Alternative technologies have been studied for the reduction of waste burden. One option for the alternative is nuclide management by the chlorination and separation of strontium, TRU (TransUranium), and rare earth elements after the thermal treatment of used nuclear fuel. The resulting products are uranium oxide and (Sr/TRU/RE) phosphate blocks. It is needed to study the evaporation behavior of the individual chloride for the development of the chloride separation process. In this study, it was investigated on the separation of the strontium chloride from uranium oxide. Vacuum distillation was used for the separation of the chlorides. The salt evaporation experiments were carried out at various temperatures. The evaporated amount of SrCl_2 was less than 0.01 g for two hours at 900°C. At 1,000°C, the evaporated amount of SrCl_2 was 0.27 g that is equivalent to the evaporation rate of less than 0.019 g/cm²-h. At 1,060°C, the weight loss was about 2.3 g after distillation, which is equivalent to less than 0.16 g/cm²-h. From the above result, it could be concluded that the operating temperature of the salt distiller should be higher than 1,060°C in order to evaporate SrCl_2 with a sufficient rate.

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Keywords: Strontium, Rare earth element, Chlorides, Vacuum distillation

Temperature Evaluation of Spent Fuel Loaded in Dried Canning Container in the PIEF Pool

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The Post Irradiation Examination Facility (PIEF) stores spent nuclear fuels that have been irradiated in a light water reactor (PWR). Most fuels are stored in the form of assemblies, while some single fuel rods are stored in a basket. Additionally, certain fuel rods have been cut for the purpose of investigating destructive PIEs. In case of damaged spent fuels, such as fuel rod cuts, additional treatment is required for their transportation and storage to contain the radioactive fission products within a defined space. To address this need, the Quiver container, introduced by Westinghouse Sweden is used for canning damaged spent fuels. This canning container has a function that removes water inside the container in the pool environment by connecting to the vacuum drying equipment. As the inside of the canning container dries, the surrounding water (which previously cooled the fuels) is drained and removed. Consequently, the temperature of the fuel increases. Although the canning container is submerged in the pool and the temperature rise is expected to be extremely limited, we conducted a temperature evaluation of the fuel to identify the process safety. The evaluated fuel conditions were determined to encompass all fuels in the PIEF with conservative considerations for burnup and cooling period. In this evaluation, the ANSI/ANS-5.1 standard was utilized to assess the decay heat from the fuel.

Keywords: Hot cell, Rod internal pressure, Fission gas, Post irradiation examination, Fuel cladding

Thermodynamic Analysis of Residual Water Evaporation in Vacuum Drying of Spent Nuclear Fuel Canisters

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Within the scholarly domain of nuclear fuel management, heightened scrutiny surrounds the enduring preservation of spent nuclear fuel, precipitating as a salient technical exigency necessitating meticulous scrutiny of fuel integrity. Of paramount concern is the imperative to ascertain the retrievability of spent nuclear fuel, as mandated by stringent regulatory frameworks exemplified by 10 CFR Part 72 in the United States. Evident in scholarly discourse is the acknowledgment of hydride re-orientation as a principal catalyst in the corrosion cascade of cladding materials, thereby accentuating the exigency of expeditiously expunging residual moisture from spent fuel canisters via vacuum drying methodologies in the preliminary treatment phase. Evaluation of drying efficacy is contingent upon adherence to prescriptive acceptance criteria delineated by the Nuclear Regulatory Commission (NRC), enshrined within the purview of ASTM C1553-16. This rubric necessitates the sustenance of vacuum pressure stability, maintaining a static pressure quotient of 3 torr for a duration exceeding 30 minutes subsequent to the cessation of vacuum pump operations. Engineering scrutiny of vacuum drying efficacy underscores dual imperatives: ensuring congruence between residual moisture levels and prognostications, or alternatively, ensuring a residual moisture quotient commensurate with safety thresholds, whilst concurrently optimizing drying efficiency to ensure economic viability. To address these imperatives, a granular examination of the thermodynamic underpinnings governing residual moisture evaporation within the vacuum drying milieu is imperative. From the vantage point of thermal engineering, discernment is warranted regarding how system parameters interface with the dynamic evolution of residual moisture states throughout decompression stages and retention intervals within canister enclosures. Our study endeavors to delineate heat transfer coefficients intrinsic to residual moisture and ascertain evaporation rates per unit area, delineating nuances across varying vacuum pump capacities, residual moisture temperature gradients, and the modulation of decompression modalities. Foreseen as seminal contributions, our research endeavors are poised to galvanize advancements in autonomous vacuum drying methodologies and engender the development of sophisticated numerical models requisite for optimizing vacuum drying protocols in forthcoming iterations.

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

Fabrication of Stabilized Pellets Using Fuel Powder Recovered From Damaged Fuel Rods by Vibratory Oxidative Decladding

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The operating conditions for the stabilized pellet fabrication process were investigated by experiments using damaged fuel rod-cut. Here, this process consists of the decladding to recover nuclear materials in the form of U_3O_8 powder from fuel rod-cut by vibratory oxidative decladding in a vertical furnace, reduction thermal treatment, milling, mixing with lubricant, pelletizing, and sintering.

The oxidation and vibration operation conditions resulting in 100% decladding efficiency and 99.95% recovery rate of U_3O_8 powder were obtained through vibratory oxidative decladding using a 200-mm long fuel rod-cut with defect such as penetrated holes in cladding tube. Through the reduction thermal treatment of U_3O_8 powder in a vertical furnace, although the reaction time is long, operating conditions with no change in particle size were secured to obtain UO_2 powder. The sound sintered pellets using simulated UO_2 powder with the burnup of 55 GWd/tU (15-year cooling) were obtained through planetary milling, mixing with 0.4wt% lubricant, pelletizing at 300 and 360 MPa, and two-step sintering at 1,300 and 1,700°C. However, the sintered density was 88% of the theoretical density. Compared to attrition milling under the same operating conditions, sinterable powder resulting in sintered pellets with above 95% theoretical density could not be produced by planetary milling.

Acknowledgements

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Keywords: Damaged fuel rod, Oxidative decladding, Vibration, Reduction, Stabilized pellet

The Radiation Dose Assessment Based on the Neutron Shield Material in Spent Fuel Transfer Casks

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The spent fuel transfer cask is an operational equipment used to safely perform a series of tasks, including withdrawing spent nuclear fuel stored in wet storage pools within nuclear power plants, loading it into empty canisters, converting the interior of the loaded canister into a dry environment, sealing it, and transferring it vertically and horizontally to storage facilities. It is designed to load the canister into the storage cylinder of a dry storage module. The upper part of the transfer cask is equipped with an openable upper lid, while the lower part of the transfer cask is fitted with a sealable lower lid. The lower lid features a bolted closure structure that can be opened by a separate mating device, allowing for the withdrawal of the canister from the transfer cask for loading into the storage cylinder of the dry storage module, and vice versa, enabling the withdrawal of the canister from the storage cylinder for loading into the transfer cask. The main body of the transfer cask consists of a cylindrical container with lead filling between inner and outer carbon steel shells. The upper and lower parts of the container are bolted with detachable lids. The circumferential side is composed of synthetic resin (NS-4-FR) for neutron shielding, carbon steel heat transfer fins for improved heat removal performance, and stainless-steel neutron shield covers. The exposed surface of the carbon steel is clad with stainless steel for corrosion resistance. In this paper, we compared and evaluated the external surface radiation dose based on the material of the neutron shield installed in the spent nuclear fuel transfer casks.

Keywords: Spent fuel transfer casks, Neutron shield, Radiation dose assessment

Mass Spectrometry Calibration for Quantification of Residual Water Inside Spent Fuel Dry Storage Canisters

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For long-term dry storage of spent nuclear fuel, it is necessary to remove the moisture inside the dry storage canister. Recent studies have shown that up to 55 mole of water remains in the container, even if it meets the drying criteria. From this research case, there is a need to re-examine the methodology for measuring residual moisture in canisters. Normally, a pressure gauge or dew point temperature sensor is used to measure the concentration of residual water inside a canister, but a mass spectrometer can be used as an instrument to measure the gas composition as well as the water vapor inside the canister. Moisture quantification studies are carried out by applying mass spectrometry to determine residual water content based on prior research conducted at Sandia National Laboratory.

In order to use a mass spectrometer for moisture quantification analysis, it must be calibrated: the peak intensity must be corrected for a specific amount of moisture in order to quantify the amount of water vapor in the sample gas. Dew point generator, reference-grade dew point sensor, air compressor, refrigerant and desiccant dryers, flow meters, and pressure gauges were used to calibrate the mass spectrometer. The compressed air passes through the dryer and reaches a dew point temperature of -40°C or less, and through a specially designed dryer down to trace levels of less than 1 ppm_v. It dries for about 8 hours or more to dry any moisture that may be present inside the piping lines that connect to the dew point sensor and mass spectrometer, respectively. Then, the dew point temperature data and mass spectrometer data are collected and calibrated according to the moisture concentration generated by the dew point generator. There was a significant difference between the reference-grade dew point sensor and the mass spectrometry measurement data, which measured the moisture concentration according to the dew point temperature, and a correction formula was calculated to compensate for this. We plan to continue additional calibration tests to verify the calculated calibration formula and sensitivity analysis tests according to the calibration environment conditions. This calibration formula is planned to be used for moisture measurement for residual moisture quantification using mass spectrometry.

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

Vacuum Drying Technologies for Dry Storage of Spent Nuclear Fuel: A Comprehensive Review

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Developed nations are increasingly turning to dry storage technology to address the saturation of on-site wet storage tanks for spent nuclear fuel. In ensuring safe dry storage, special attention must be given to residual water inside the canister. If left unattended, residual water can lead to chemical corrosion and integrity issues during prolonged storage. The drying standards of U.S (NUREG-1536 and SFST-ISG-11) dictate that residual water must not exceed the saturated water vapor pressure (below 3 Torr for a duration of 30 minutes). The majority of technical data regarding spent nuclear fuel drying equipment, particularly from overseas entities like Holtec International, remains undisclosed to the public. Furthermore, the prospect of engaging in active technological exchange concerning core technologies is expected to be challenging. Hence, Korea's domestic academic sector faces the imperative of developing its proprietary technology. This involves the advancement of vacuum drying technology aimed at achieving three key objectives: (1) minimizing residual water, (2) precisely determining moisture levels, and (3) optimizing drying duration. Based on these requirements, this study delves into previous research pertaining to existing vacuum drying methods for spent nuclear fuel, as well as the quantitative evaluation of residual water. It compiles the technological development history, compares various technologies assessed in literature, and evaluates their disparities. Methodologically, the study is broadly divided into three approaches: multi-physics simulation considering multidimensional characteristics, formulation development based on theoretical modeling, and direct measurement of residual water through experimental verification. Examples of recent experimental studies include reports from the Savannah River National Laboratory and the Idaho National Laboratory. These reports provide valuable insights into predicting residual water and corresponding experimental findings essential for impact assessment. In conclusion of this review process, based on earlier research summaries, the study's authors identified key test variables influencing the evaporation process during vacuum drying. We organized conducted tests according to parameters such as the number of decompression steps, hold time, vacuum pump capacity, and residual water temperature, while also providing insights into technological blind spots necessitating future development.

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Keywords: Vacuum drying, Spent nuclear fuel, Dry storage system, Residual water, Evaporation

Long-Term Integrity Verification of BSS (Borated Stainless Steel) Neutron Absorber

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The spent nuclear fuel generated during the operation of NPP (nuclear power plant) is stored in a wet storage pool in NPP. Currently, in Korea, a high-density storage rack is used to store spent nuclear fuel in order to increase the storage capacity of spent nuclear fuel, and neutron absorbers are used in the high-density storage rack to control the criticality of spent nuclear fuel. In general, Boral or Metamic neutron absorbers are installed on the wall of the high-density storage rack, but BSS itself serves as a neutron absorber and a structural material.

In this paper, a corrosion acceleration test was performed to verify the long-term integrity of BSS. Since the corrosion rate of a general alloy follows the Arrhenius' equation as a function of temperature, the experimental temperature was selected based on this equation. Since SS304, the main component of BSS, produces a hematite or magnetite oxide film according to the DO concentration, the corrosion acceleration experimental conditions were selected by deriving an appropriate DO concentration. Corrosion acceleration tests were conducted for two years to take into account the wet storage conditions for 60 years. In addition, a neutron attenuation test was performed before and after the corrosion acceleration test to evaluate the absorption capacity of the neutron absorber, and a neutron attenuation test was performed at each of the five points of the BSS specimen.

As a result of the test, there was a difference of 0.43% for the 12-month corrosion acceleration test, 1.63% for the 18-month specimen, and 0.33% for the 24-month specimen, but it was confirmed that there was no effect on safety because it was within 5% of the technical specification in the nuclear power plant operation procedure. From these results, it was evaluated that the deterioration of BSS in the spent nuclear fuel wet storage pool would not have a significant effect in terms of neutron absorption capacity.

Keywords: Spent fuel, BSS, Corrosion, Neutron attenuation test

Review of the NWTRB Report on the Disposal Evaluation of Spent Nuclear Fuel Stored in DOE's Dual-Purpose Canisters

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Currently, the United States stores spent nuclear fuel at over 70 sites, with the majority being stored in canisters designed for storage and transportation. These canisters, mostly welded dual-purpose canisters not designed for disposal, are being replaced by larger-capacity canisters. The U.S. Department of Energy (DOE) actively conducts these studies, and the U.S. Nuclear Waste Technical Review Board (NWTRB) has issued a report after reviewing the DOE's research and development activities related to the disposition of spent nuclear fuel stored in dry storage casks at independent spent fuel storage installations (ISFSIs). The Board examined three alternative approaches for managing commercial spent fuel stored in DPCs: storing spent fuel at ISFSIs indefinitely, repackaging the spent fuel into smaller canisters prior to disposal in a geologic repository, and direct disposal of spent fuel in DPCs in a repository. This report discusses the implications of each alternative and presents the Board's observations, findings, and recommendations regarding the alternatives. Therefore, this paper reviews the report published by NWTRB.

Keywords: NWTRB, DOE, Spent fuel, DPC

Preliminary Study of Cladding Integrity of Spent Fuel Using Probabilistic Safety Assessment

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This study aims to verify the integrity and safety of spent nuclear fuel (SNF) cladding using a probabilistic safety assessment (PSA) based on Python code. The specific objective is to estimate the number of each size cracks in the cladding using probability density functions (PDFs) and to classify the fuel rods as intact, undamaged, or damaged.

The crack size are crucial factors in evaluating cladding failure under external stress. Therefore, this study can serve as a fundamental database for probabilistic safety assessment. Furthermore, the proposed method can be applied to vacuum-dried SNF, which has different crack distributions depending on its operational history (temperature, fluence, and storage duration).

The coding in this study was based on the SNL report SAND90-2406, which presents a pioneering probabilistic safety assessment methodology. Future studies will involve customizing the model to consider domestic storage conditions and utilize the developed framework to distinguish between intact, undamaged, and damaged fuel rods.

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Keywords: Spent nuclear fuel, Cladding integrity, Probabilistic safety assessment, Crack, Crack distribution

Structural Analysis of SNF Cladding Due to Corrosion-Induced Swelling of Pellet Under Residual Water Environment

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This study investigates the integrity and safety of spent nuclear fuel (SNF) cladding under pellet swelling conditions caused by residual water using finite element method (FEM) simulations in ANSYS. The primary objective is to assess whether the cladding maintains its structural integrity under swelling conditions with pre-existing cracks.

The fracture toughness (K_{Ic}) of the cladding was evaluated to provide an objective measure of its resistance to crack propagation. A fixed and displacement boundary condition was applied at both ends of the FEM model, and a 36% radial swelling of the pellet was simulated without axial swelling, resulting in a 16% increase in diameter. The cladding length was assumed to be 15 cm with axial symmetry, and the crack is located 1 mm away from the space grid.

A preliminary analysis was conducted without considering nonlinear contact to measure the cladding's fracture toughness. The result, 92 MPa, exceeds the K_{Ic} criterion of 5 MPa for DHC conditions, indicating that residual water-induced corrosion promotes crack growth.

Future studies will investigate the safety of cladding under accident scenarios by considering additional factors such as bend or pinch load.

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Keywords: Spent nuclear fuel, Cladding integrity, Fracture toughness, Swelling

Selection of Specimens for Evaluating Concrete Deterioration Properties in Spent Nuclear Fuel Dry Storage Facilities

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Concrete structures in spent nuclear fuel interim storage facilities should maintain their shielding ability and structural integrity under normal, off-normal, and accidental conditions. Dry storage systems for spent nuclear fuel have been in use for more than 20 years, and the period of use is expected to further increase. Therefore, concrete structures may deteriorate if an interim storage facility operates for several decades. Even in the event of deterioration, it is crucial for concrete structures to retain their functions, such as radiation shielding and structural integrity. Therefore, establishing an analysis methodology capable of evaluating whether deteriorated concrete structures maintain their integrity under normal or off-normal conditions and during accidents is necessary.

We plan to conduct both static and dynamic material property tests on concrete that meets the KEPIC SG and SNC requirements. After introducing a crack in the concrete, we will evaluate the impact on its shielding capability based on the size of the crack. Moreover, we plan to conduct experiments to evaluate how the static and dynamic properties of concrete change when exposed to fire. In this study, the specimens for these tests have been selected.

In the case of dynamic physical property testing, a specimen with a diameter of 100 mm and a length of 200 mm will be manufactured, and then specimens with a diameter of 40 mm and a length of 50 mm with an internal hole diameter of 5 mm will be produced by waterjet processing. To evaluate the effects of concrete cracking, specimens the size of concrete flexural strength test specimens (100×100×400 mm) will be used. Certain specimens will be manufactured with a 5 mm diameter rebar positioned at the bottom to ensure they retain their shape in the event of cracking. The manufactured specimens will be subjected to a three-point bending test using a universal testing machine to generate cracks. Specimens for fire testing are planned to be manufactured with a diameter of 800 mm, height of 450 mm, and internal diameter of 100 mm, and will be cored after the test to extract specimens for physical property testing.

The results of material testing with the manufactured specimens will be used for structural analysis considering the material properties of deteriorated concrete.

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Keywords: Dry cask storage system, Interim storage facility, Concrete deterioration, Dynamic material testing

Spent Fuel Handling Guidelines at Nuclear Power Plants

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This study provides technical information for the spent fuel handling guidelines of the fuel handling equipment at nuclear power plants. These guidelines involve general instructions, precautions, and limitations to be observed during the fuel handling to prevent fuel damage and to ensure a radiological protection, adequate cooling and a prevention of a criticality. ANSI/ANS-57.1 shows the main interlock design requirements of the fuel handling equipment.

The RM (Refueling Machine) (or SFHM, Spent Fuel Handling Machine) hoist is designed such that the operator would not be exposed to a radiation dose rate greater than 2.5 mrem/hr by using up-limit interlock at the reactor and the FTS (Fuel Transfer System) upender in each building. The RM hoist speeds when seating grapple onto a fuel, inserting or withdrawing a fuel into the FTS upender and the reactor should be set based on the fuel handling guidelines. The RM has capability to determine off-index operation during fuel loading and off-loading at reactor. The off-index is defined as a fuel assembly moving diagonally away from adjacent fuel assemblies at reactor. This then allows the hoist to operate at high speed under allowable acceleration and deceleration to a position above full down. This off-index operation gives advantages to get reduced hoisting time and to protect space grids from interference during hoist operation at reactor.

Lastly, the FTS upender and RM (or SFHM) should be interlocked to ensure proper equipment position during each phase of transfer operations to prevent damage to the fuel assembly. The spent fuel seating surface in the FTS upender should be designed to be same as the elevation of the SFSR (Spent Fuel Storage Rack), the cask in the cask loading pit and the NFE (New Fuel Elevator) for efficient operation of the SFHM hoist. This study will be beneficial guidelines for spent fuel handling in various fuel handling process.

Keywords: Spent fuel handling, Fuel handling equipment, Spent fuel handling machine, Refueling equipment

Establishment of Isotope Inventory DB to Develop Spent Fuel Burn-up Calculation Algorithm

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Enhancing the safety of long-term spent fuel management is a critical aspect of nuclear power plant operation. Cross-validating the burn-up through direct measurements in the transfer process of spent fuel is an important step in ensuring the integrity and safety of the disposal process.

The Nuclear Regulatory Commission (NRC) recommends physical burn-up measurements in the transport, storage, and management of spent nuclear fuel. By directly measuring burn-up during the transfer process, operators can verify and validate the information provided by the nuclear fuel management system.

Cross-validation involves comparing different methods or sources of data to ensure consistency and accuracy. In the context of spent fuel management, cross-validating burn-up measurements means comparing the results obtained from direct measurements with those calculated or estimated through other means, such as reactor operation records or simulations.

There are various methods to estimate the burn-up of spent nuclear fuel indirectly, given the limitations of direct chemical analysis methods. Instead, non-destructive indirect methods utilizing nuclear fuel characteristics are being explored. However direct chemical analysis method are time-consuming and costly. Consequently, estimating the burn-up of stored spent nuclear fuel using such direct methods is practically unfeasible. Instead, non-destructive indirect methods utilizing nuclear fuel characteristics are being explored. The approach involves establishing an isotope inventory database and developing an algorithm for estimating burn-up based on various parameters such as fuel type, initial enrichment, burn-up, and cooling time. To achieve this, depletion calculations were performed to determine the burn-up, considering the aforementioned parameters. To achieve this, depletion calculations were performed to determine the burn-up, considering the aforementioned parameters. Through this process, a database reflecting various characteristics of spent fuel was established. It was confirmed that the ratio of burn-up to nuclide inventory can be expressed as a unique linear function depending on the cooling time and initial enrichment. The next steps involve further research to understand the relationship between the undetermined coefficients of the linear function and the specific characteristics of spent fuel. This study will enable the development of a robust burn-up estimation algorithm. In summary and present a promising approach to indirectly estimate burn-up of spent nuclear fuel, offering a more efficient and cost-effective alternative to direct chemical analysis methods.

Acknowledgements

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Keywords: Burn-up, Data base, Spent fuel, Inventory, Depletion calculation

ISO 24389 Series and Its Utilization for Radioactive Waste Data Analysis Framework

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The WG within ISO TC85 SC5 (Nuclear installations, processes and technologies) is as described as below, and WG5 is in charge of standards for the management of radioactive waste generated from nuclear facilities. MIRAE Co., Ltd has led the development and published ISO 24389 Part-1 - General principles, objective and practical approaches in 2023.

1. CAG: Chair's Advisory Group
2. WG1: Analytical methodology in the nuclear fuel cycle
3. WG4: Transportation of radioactive material
4. WG5: Characterization and waste management
5. WG8: Nuclear criticality safety
6. WG13: Decommissioning(decontamination, dismantling and remediation)

Through the preparing ISO 24389 Part-1, WG members agree that the remainder of issues(Part-2 through part-7) that is originally proposed by Korea are going to be developed as future works. The Project Lead for Part-2 and Part-7 was determined.

1. Part 2(ISO 24389-2): Pre-disposal (*will be led by Korea*)
2. Part 3(ISO 24389-3): Disposal (To be determined)
3. Part 4(ISO 24389-4): Governance (To be determined)
4. Part 5(ISO 24389-5): Data quality (*will be led by UK*)
5. Part 6(ISO 24389-6): Documentation (To be determined)
6. Part 7(ISO 24389-7): Supporting systems (To be determined)

This paper will provide the brief of ISO 23489 series and its utilization for bigdata analysis framework, which could be used for conceptual basis of radioactive waste data science and AI-oriented modeling.

Acknowledgements

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Keywords: Radioactive waste, Publication, Nuclear facilities, Management, Standard

Dissolution and Precipitation Behaviors of Barium Oxide in Molten Salts

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The utilization of molten salt to separate highly radioactive nuclides from spent nuclear fuel, subsequently precipitating them into solids, is garnering attention as a means to reduce the disposal area of spent nuclear fuel. Our previous work has demonstrated the efficient separation of Sr, a highly radioactive nuclide, from simulated nuclear fuel using molten MgCl_2 and LiCl-CaCl_2 salts. While Ba, although not highly radioactive like Sr, shares similar characteristics as a group 2 alkaline earth metal element. Therefore, if Sr can be effectively separated using molten salts, it is likely that Ba can be as well. However, the behavior of Ba in the molten salts utilized for Sr separation remains unexplored. In this investigation, we examine the reaction of BaO with various salts (i.e., LiCl , LiCl-CaCl_2 , CaCl_2 , and MgCl_2) by immersing BaO reagent in the respective salts and analyzing both the salt solution and the residual solid. Additionally, a precipitant (K_2CO_3) is introduced to the salt resulting from the reaction of BaO with molten salts to examine the precipitation of Ba as a solid.

Acknowledgements

This study was supported by the National Research Foundation of Korea grant funded by the Korean government (No. 2021M2E3A1040059).

Keywords: Molten salt, Separation, Barium, Spent nuclear fuel, Precipitation

Layout Planning of Automated Pyroprocess Equipment in Hot Cells Based on the Relationship Between Equipment

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The integrated pyroprocess is a batch process in which the baskets containing nuclear materials are transferred from one equipment to the next. In normal operations, since the order in which the process basket is transferred is determined according to the process operation procedure, the transfer efficiency, or the process efficiency can be improved by reducing the distance between the consecutive equipment.

However, in the case of manual operation, it is recommended that the equipment in the hot cell is placed near the shielding window to secure the operator's field of view and to ensure remote operability. Therefore there are limits to improving the transfer efficiency by adjusting the equipment layout. On the other hand, in automated operation, process baskets or electrodes are transferred by an automated handling device which can reach most areas of the hot cell, so there are fewer spatial restrictions in arranging the equipment, which can further increase transfer efficiency and reduce the area compared to manual operation for the same process.

In this paper, the required equipment for automated process operation were listed and the relationship between equipment was defined by reflecting the operating procedures and the importance or frequency of the transferred baskets or process materials. Through the relationship, the total closeness was calculated and the layouts can be derived in which the process devices with higher relationship were placed adjacent to each other.

Acknowledgements

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Keywords: Pyroprocessing, Integrated process automation, Layout planning, Hot cell

Specific Heat Measurement With Differential Scanning Calorimeter

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

Specific heat is a property that describes the ability to absorb and retain heat from the outside. In addition, the specific heat of nuclear fuel is one of the most important properties along with thermal expansion and thermal conductivity. In the nuclear field, the measurement of specific heat is one of the most important factors in the characterization of materials after irradiation in a reactor.

There are several ways to measure specific heat, and this article will focus on measuring specific heat using a Differential Scanning Calorimeter (DSC). The DSC measurement method uses Indium, Tin, and Zinc to create temperature calibration and sensitivity calibration files. Based on the temperature and sensitivity calibrated file, the baseline is measured with an Alumina pan. Then, the SRM material, sapphire, is placed in the alumina pan and the standard data is measured. Finally, add the sample material you want to measure to the Alumina pan and measure the sample data. After that, you can load standard data and sample data into the program to get a Cp graph.

It is believed that the specific heat measurement with DSC developed in this study will be helpful for increasing reliability and stability evaluation of nuclear fuel and spent fuel.

Keywords: Differential scanning calorimeter (DSC), Standard reference material (SRM), Standard reference data (SRD), Specific heat, Nuclear fuel and materials data center

A Study on the Long-Term Cooling Effect of Hydride Reorientation in Unirradiated Zircaloy-4 Cladding (12, 18 & 24 Months)

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The hydride reorientation (HR) in spent fuel cladding has an impact on the health of spent fuel. During dry storage drying, an interim storage method, dissolved hydrides in the circumferential direction are rearranged into radial hydrides in the radial direction. The factors that affect the hydride reorientation are hoop stress, peak temperature, cooling rate during storage, and hydrogen concentration deposited during combustion, which increases the brittleness of the spent fuel cladding. This study evaluated the hydride reorientation behavior of unirradiated Zircaloy-4 (CWSRA) cladding tubes loaded with hydrogen under long-term cooling conditions (12, 18, and 24 months) simulating a 90 MPa hoop stress condition.

The analysis of the offset strain (%) by hydrogen concentration in long-term cooling showed that the longer the cooling period, the higher the brittleness. In addition, the specimen with lower hydrogen concentration showed a larger vertical and radial hydride generation than the specimen with higher hydrogen concentration.

In the hydride reorientation test of the irradiated cladding tube, the brittleness of the offset strain was stronger than 2%, and the relatively high brittleness of the specimen with low hydrogen concentration was confirmed. In actual spent fuel cladding, it is believed that neutron irradiation hardening and dislocation loop transitions facilitate the reorientation of dissolved hydrogen into radial hydrides. To quantify these results, it is necessary to confirm their reproducibility through repeated tests.

Acknowledgements

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

Development of CaO-based Adsorbents for $^{14}\text{CO}_2$ Capture

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Pyroprocessing is one of the methods for converting spent nuclear fuel accumulated within nuclear power plants into reusable nuclear fuel. However, gaseous fission products (FPs) such as Cs-137, Tc-99, and I-129 are generated during the pyroprocessing process. Among these FPs, C-14 has a long half-life (5,730 year) and it can easily be absorbed to human body. In order to prevent exposure of C-14 to environments, it is essential to strictly manage C-14 emission.

In the oxidation process of spent nuclear fuel in pyroprocessing, it is known that C-14 is converted to a gaseous $^{14}\text{CO}_2$ in the presence of oxygen during heat treatment of the spent fuel. Among the commercial CO_2 adsorbents, CaO is generally used because of its low cost and excellent adsorption performance.

In this study, the CO_2 capture behaviors were investigated using three different CaO-based adsorbents with distinct pore characteristics. The greatest increase in mass due to CO_2 capture was measured at 600°C for all samples. The cyclic stability of CaO adsorbents for CO_2 capture was evaluated through 10 adsorption-desorption tests using TGA.

Keywords: Pyroprocessing, Spent nuclear fuel, C-14, Fission product, CaO

Annular Metal Fuel Fabrication Technology Development Using Injection Casting Method

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Advanced Small Modular Reactors (SMRs) which employ non-light water coolants are currently being researched for their various advantages including inherent safety, nuclear non-proliferation, economy, fuel utilization, load-following capabilities, and long-core life. Metal fuels, one of the advanced small reactor candidate fuels, offer superior safety due to their high thermal conductivity that maintains lower center temperatures. However, high swelling characteristic of metal fuels leads low smeared density about 55%. Moreover, the low melting temperature of metal fuel requires sodium filling in the fuel-cladding gap to ensure thermal margin. This not only leads to a decrease in yield after manufacturing but also complicates the storage and reprocessing of spent nuclear fuel. Annular fuel design facilitates maintaining the smear density without sodium filling, which simplifies the manufacturing process and eases the treatment of spent nuclear fuel.

In this study, the feasibility of using injection casting to manufacture annular metal fuel with a composition of U-10wt%Zr was examined. The injection casting method applied various designs of annular molds and process parameters. After casting, the fuel core was successfully immersed in a 10 M NaOH solution to remove the internal quartz mold without damaging the metal fuel core, finally producing an annular metal fuel core with an outer diameter of 6.40 mm, inner diameter of 3.00 mm, and a length exceeding 60 mm. The manufactured fuel core underwent casting defect evaluation, visual inspection, density measurement, microstructure, and phase analysis to review the feasibility of each manufacturing methodology and process parameter.

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Keywords: SMR, Metal fuel, Annular fuel, Injection casting, Uranium-zirconium alloy

Capture and Release of Oxygen in $\text{YBaCo}_4\text{O}_{7+x}$

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Temperature swing adsorption (TSA), which does not require purge gases for regeneration, has advantages for oxygen removal in compact systems. Materials for oxygen capture by the TSA process must reversibly incorporate/release oxygen into/from their matrix by controlling temperature. Especially, $\text{YBaCo}_4\text{O}_{7+x}$ could control its oxygen contents at relative low temperature ($\sim 350^\circ\text{C}$). For use in low oxygen environment, it would be particularly important to design high-oxygen-intake/release-rate materials. To tune the rates, understanding the oxygen-intake/release behavior must be accomplished ahead.

In this work, $\text{YBaCo}_4\text{O}_{7+x}$ was synthesized by solid state reaction. Oxygen intake and release by TSA were observed thermogravimetrically. The physical and chemical properties were characterized and compared before/after oxygen intake.

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Keywords: Oxygen storage materials, Oxygen absorber, $\text{YBaCo}_4\text{O}_{7+x}$

A GoldSim Model for Mass Flow Analysis and Economic Evaluation of Back-End Fuel Cycle Incorporating Radionuclide Management Process

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KAERI is developing radionuclide management processes as a conditioning process aimed at reducing the environmental burden associated with spent fuel disposal. This process involves managing radionuclides with similar properties by introducing various technological options capable of separating and recovering radionuclides from spent fuels. Specifically, it targets the management of high-heat, high-mobility, and high-toxicity radionuclides, which can significantly impact the performance of the disposal system.

In previous studies, potential process flows for the radionuclide management process were screened out based on feasibility among various options. And a GoldSim model was established to calculate the mass flows for these potential processes. In this study, the developed model was integrated with the mass flow models for other back-end fuel cycle facilities to complete a system-level mass flow model. The system-level mass flow model enables the derivation of mass flows for the fuel cycle facilities, reflecting the transition scenarios based on Korean nuclear plans. And an economic evaluation model, which uses resulted mass flows, was established as part of the integrated evaluation framework, and a preliminary economic evaluation was conducted for verification purposes.

In the future, the economic evaluation model will be utilized to quantitatively evaluate fuel cycle system options alongside effectiveness and safety evaluation models. Ultimately, quantitative evaluation indicators will be derived based on finalized input data and scenario, leading to the proposal of an optimal nuclear cycle option.

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Keywords: Radionuclide management process, Performance evaluation model, GoldSim model, Back-end fuel cycle, Mass flow analysis, Economic evaluation

Reactor Design for Recovering the Metal Phosphates Precipitate by Segregation of the Salt Ingot With an Aid of a Dividing Plate Structure and Tapered Crucible

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A phosphorylation reaction rate is very fast and the metal phosphates as a reaction product precipitated on the bottom of the molten salt crucible. One of the recovery methods of the metal phosphate precipitates is segregation the lower part (precipitates) of the salt ingot using the various cutting tools. In KAERI's previous phosphate precipitation tests, the dispersed powder of lithium phosphate (Li_3PO_4) as a precipitation agent reacted with various metal chlorides in a simulated LiCl-KCl molten salt. The reaction of metal chlorides composed of actinides such as uranium and three rare earths (Nd, Ce and La) with lithium phosphate is a solid-liquid reaction. In this study, the precipitation reactor design (a tapered crucible with polished inner surface) used for phosphorylation reaction showed that the salt ingot with metal phosphate precipitates could be detached from a tapered stainless steel crucible. We proposed that the recovery of precipitates from a salt ingot was possible by introducing a dividing plate structure into a molten salt and by positioning it at the interface between salt and precipitated metal phosphate. A series of experiments using three kinds of the dividing plate structure carried out in a bench scale reactor. Experimental results shows a good performance. In order to increase the effect of stirring a molten salt, some baffles install on the inner wall of a scaled-up precipitation reactor. Considering inserting the dividing plate structure that segregates between salt ingot and precipitates into a molten salt in the reactor crucible the baffles installed in a reactor (tapered crucible) can be an obstacle. A concept of detachable baffle introduced to avoid a structural conflict of the baffles and the dividing plate structure. In order to make easy handling large size of a salt ingot according to a scaled-up reactor the dividing plate for precipitates can be stable position horizontally by installing an upper dividing plate for segregating a salt ingot in two pieces.

Acknowledgements

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

Applications of the Oxidation Treatment Method for the SF Volume Reduction in the SNF Dry Storage Facility

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Oxide Treatment Technology Currently, oxide treatment technology in Korea was conducted as part of DUPIC research led by the Nuclear Research Institute in the 1990s. Various studies have been conducted abroad as part of the AIROX process. However, rather than improving the storage efficiency of spent nuclear fuel, the above research and development focused on the recycling of U-235 remaining in the spent nuclear fuel. Rather than improving the storage efficiency, research and development were conducted with the aim of recycling spent nuclear fuel using simple thermal and mechanical methods without nuclear species separation.

On the other hand, oxide treatment technology aimed at improving the storage and disposal efficiency of spent nuclear fuel should be designed with a focus on improving the storage and disposal efficiency, and it is necessary to secure the properties of the oxide blocks and waste generated thereafter and review the connection with storage and disposal thereafter. Therefore, oxide treatment technology needs to be studied in the following technical aspects.

Gas generated during the mechanical low interest process Solid waste treatment method. As a result, highly heat dissipating nuclides such as Cs contained in spent nuclear fuel volatilize, and solid waste constituting the aggregate is also inevitably generated, so it is necessary to study the inside of the treatment bread.

Various preceding studies have been conducted on the technology for collecting high heat dissipating nuclides such as Cs in Korea, and it is expected to be important to study the gas waste collection method generated in oxide treatment technology by referring to a little research Quantitative analysis is expected to be necessary to prevent the problems with the fin ventral tube and the problems that will occur during the powdering process. About 35% of the volume of the crude nuclear fuel must also be considered. Predicting the volume change through phase change modeling is also expected to be a method.

The optimal oxide body block volume considering the volume change must be derived, and the effects on various temperatures, nuclear thresholds, and radiation in the process must be derived.

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

Restoration of KOFA Spent Fuel Assembly

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The 16×16 Korea Optimized Fuel Assembly (KOFA) stored in the Post-Irradiation Examination Facility (PIEF) was transported from a nuclear power plant to PIEF in 1993 and has since been stored in the fuel storage pool. The top nozzle of the fuel assembly was dismantled, and tens of fuel rods, including intact and damaged fuel rods were extracted from the fuel assembly for examining the fuel characteristics. The remaining fuel rods are currently stored in the assembly. To handle and transport the dismantled fuel assembly in the near future, it is necessary to restore a top nozzle and ensure the mechanical integrity of the fuel assembly.

Previously, we developed a reassembling and restoration method by designing and bolts and handling tools for the KOFA spent fuel assembly. In this research, we restore the KOFA fuel assembly through the following steps: aligning of top nozzle and guide tube (1st), bolt screwing (2nd), bolt fastening (3rd), and plastic deformation of the bolt wing (4th). Our results on the restoration of the top nozzle for KOFA are expected to be utilized for the development of a restoration method for other fuel assembly types.

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

Development of Large Capacity CANDU Spent Nuclear Fuel Transport Cask (KTC-360)

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As of January 2024, the Wolseong Nuclear Power Plant (NPP) is temporarily storing more than 500,000 bundles of CANDU spent nuclear fuel (SNF) in wet storage pools and dry storage facilities. Wolseong NPP developed HI-STAR 63 casks for transportation of SNF from wet storage to dry storage facilities and has been using them since 2010. The capacity of the HI-STAR 63 cask is 120 bundles (2 baskets). Since the transport capacity of 120 bundles is not sufficient enough, it was necessary to develop a large-capacity transport cask to transport SNF to interim storage or disposal facilities. There was also a need to develop a larger capacity cask for transport within the Wolseong NPP. Once the interim storage facility is established, the SNF temporarily stored at the NPP site will be transferred to the interim storage facility sequentially. CANDU SNF will be transported to the interim storage facility before PWR SNF. The absolute number of bundles for CANDU SNF is higher than for PWR SNF.

The KAERI, KONES, and MOOJIN have developed the KTC-360 (Korean Transportation Cask-360), a large-capacity transport cask that can transport 360 bundles of CANDU SNF at a time. By analyzing the stored SNF of the Wolseong Dry Storage Facility, the burnup of design basis fuel was selected with an average burnup of 7,800 MWD/MTU. The minimum cooling period for the design basis fuel was selected as 6 years based on the analysis of the operating period of the Wolseong NPP and the SNF transport scenario. The carrying capacity of the new KTC-360 cask was selected to be 360 bundles of SNF in six fuel baskets, taking into account the topographical characteristics of the Wolseong NPP and road conditions.

The main components of the new transport cask are made of forged carbon steel, stainless steel, and shock absorbing materials (wood). The main dimensions of the transport cask are 2,604 mm × 4,322 mm × 3,563 mm (H), and the total weight is approximately 69 tonnes. The mechanism of the connection parts such as the lid and carrier of the cask was designed to maintain the compatibility with the existing storage facilities and basket handling equipment of Wolseong NPP. Upper and lower shock absorbers were attached to the top and bottom of the cask to absorb the load on the cask in the event of a drop or collision.

The design was approved by the regulatory agency after a long development process of six years, including safety analysis, safety evaluation test, and manufacturability verification. During the design approval process, it demonstrated excellent performance and safety even under harsh conditions such as drops, fires, and water immersion.

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Keywords: KTC-360, Transport cask, CANDU SNF, Interim storage facility, Safety evaluation test

Comparison of Required Auxiliary Equipment for Concrete Overpack System

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To select the method of dry storage for SNF in consideration of worker's ALARA, it is crucial to establish a unified operational process by method of storage to compare the exposure dose. However, even for the same operation method, the operational process differs depending on auxiliary equipment, transportation means etc. considered by system. Concrete overpack, one of the dry storage methods, has been operated by various systems and auxiliary equipment considered vary. To derive a unified operational process for the concrete overpack, it is necessary to analyze the auxiliary equipment commonly used. This study analyzed the major auxiliary equipment considered by concrete overpack used in foreign countries.

Concrete overpack currently used includes HI-STORM 100, NAC-UMS, and FuelSolutions system. Those systems were designed by HOLTEC, NAC, and EnergySolutions, respectively. Each of the system has submitted a FSAR for the approval. Within these reports, the main auxiliary equipment used in each system is specified. Therefore, this study analyzed the major auxiliary equipment considered for each system through the FSAR and derive the auxiliary equipment considered commonly. The auxiliary equipment considered commonly were inlet and outlet vent screens, temperature element, and alignment device. To prevent introducing debris from outside, Inlet and outlet vent screens were considered commonly. Temperature element was considered commonly to determine whether the SNF in overpack is intacted. To transfer canister to transfer casks from overpack, alignment device was considered commonly. As inlet and outlet vent screens and temperature element were considered commonly, it is considered that the installation and removal of those equipment should be considered when deriving a unified operation process for the concrete overpack. In addition, it is considered that the operation of transferring canister should include the installation of alignment devices before and after the operation.

This study analyzed the auxiliary equipment specified in the FSAR of concrete overpack system. The auxiliary equipment considered vary, but it was found that inlet and outlet vent screens, temperature element, and alignment devices were considered commonly. The results of this study are expected to be utilized in the future to derive a unified operational process for concrete overpack.

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Keywords: Spent nuclear fuel, Concrete overpack, Auxiliary equipment, Operational process

~~Review on TLAA for Boron Depletion of Neutron Poison in Foreign Dry Storage Systems of Spent Nuclear Fuel~~

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Withdrawal

Estimation of Failure Parameters of RCT Simulation Model of Spent Nuclear Fuel Cladding Using Metamodel

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Maintaining cladding integrity is important for safe and economic spent fuel management. High burnup spent nuclear fuel has zirconium-hydrides, reoriented hydrides, and other forms of mechanical defects. The fracture resistance of cladding varies depending on the Radial Hydride Continuity Factor (RHCF). In particular, a high RHCF significantly reduces the fracture resistance to Mode-3 failure under pinch load. The mechanical behavior of cladding with reoriented hydrides under pinch load can be evaluated using the Ring Compression Test (RCT). Chan et al. demonstrated that the load required for void formation and crack initiation at the interface of Zr matrix and hydride is significantly lower than those for hydride and Zr matrix. This study proposes a fracture parameter estimation method for cladding with reoriented hydrides to scientifically and systematically evaluate the fracture resistance of cladding under pinch load.

A finite element model based on continuum damage mechanics and a ductile damage model has been developed to evaluate the fracture resistance of cladding by RCT simulation. This model takes into account the complex interactions between the Zr matrix, the hydrides, and the interface under the stress field generated by RCT. However, determining the fracture parameters of the Zr matrix, hydrides, and interface is challenging mainly due to the lack of experimental data. The fracture parameters selected in the ductile damage model were the equivalent strain values according to stress triaxiality related to damage initiation and the fracture energy value related to damage evolution. The equivalent strain parameters were selected by considering failure under three stress triaxialities: pure shear, uniaxial tension, and plane strain. The RCT simulation model is computationally demanding and calibration of fracture parameters using optimization with the model is prohibitive. In this study, a parameter estimation framework using metamodeling technique is proposed. For metamodeling, the design of experiments data set was generated. The range of appropriate parameter values and the effects of these parameters were analyzed using ANOVA (Analysis of Variance). The generated data set was used to construct a kriging metamodel. The kriging model, a response surface model that uses a stochastic process to express the relationship between the objective function and the design variables, can account for the uncertainty associated with hydride reorientation. Using the kriging metamodel, optimization was performed to estimate fracture parameters based on RCT force-displacement curves. The purpose of this study is to evaluate cladding failure under pinch load by evaluating the effects of RHCF on the reduction of fracture resistance.

Acknowledgements

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Keywords: Spent nuclear fuel, Hydride reorientation, Pinch load, SNF transportation, Crack propagation, Parameter estimation

Standardization Certification for SNF Burnup Calculation Methodology

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For the safe disposal and interim storage of spent nuclear fuel (SNF), operators must provide crucial information regarding parameters such as burn-up. Burn-up denotes the energy extracted from nuclear fuel during its irradiation in a reactor, playing a pivotal role in assessing the condition and potential hazards associated with spent nuclear fuel. Directly measuring burn-up during the transfer process enables operators to verify and validate information from the nuclear fuel management system.

Achieving this requires both the development of specialized equipment capable of accurately measuring burnup and an sophisticated algorithm for calculating accurate combustion volume from measured data, accompanied by a standardized certification procedure to ensure technology reliability.

This equipment encompasses various components, including detectors, mobile devices, and auxiliary facilities, each requiring specific standards to streamline the standardization process. However, establishing comprehensive standards for all components presents significant challenges. In this paper, standardization of the burn-up calculation process was identified as a key element of standardization related to burn-up of SNF, and a strategic plan was developed accordingly.

The formulation of the standard certification table of contents must be meticulously crafted, reflecting a systematic approach to calculating burnup by correlating each factor with the requisite elements of the evolving methodology for measuring burn-up. This methodology ensures a robust framework for certifying burnup calculation processes, enhancing the overall reliability and efficacy of SNF management practices.

By implementing the outlined standardization plan in this research, it is expected to reduce uncertainties in domestically produced systems for estimating burn-up, ensuring reliability and expanding the applicability of nuclear power plants.

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Keywords: Spent nuclear fuel, Burnup, Standardization, Methodology

A Study on Characteristics of Ballooning Rupture of Spent Fuel Cladding Under Accident Condition

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In the event of a loss of cooling function due to a failure of a cooling water pump, or a loss of cooling water due to an ultra-large-scale earthquake or an accident due to sabotage, the spent nuclear fuel pool, which is a facility on the site that stores spent nuclear fuel in a nuclear power plant, is subjected to high pressure due to the accumulation of radioactive materials such as Xe, Kr, I, and Cs gases in the Zircaloy-4 cladding tube, and if the temperature rises above a certain temperature due to hoop tensile stress, damage will occur. In other words, damage to the Zircaloy-4 metal cladding can lead to a massive release of radioactive materials, especially gaseous and volatile radioactive materials, into the atmosphere if the accident expands. As a result of this study, which simulates a serious accident in the pool of spent nuclear fuel rather than the results of the previous study that simulated a serious accident in a reactor, it was confirmed that ballooning of a Zircaloy-4 metal cladding tube is possible even at a relatively low internal pressure in the Zircaloy-4 metal cladding tube. This study aims to present the ballooning failure behavior and failure criteria of a Zircaloy-4 metal cladding tube by comparing them with the results of previous studies that simulated a serious accident in a reactor.

Keywords: Spent fuel pool, Zircaloy-4 spent fuel cladding, Ballooning, Loss-of-coolant-accident

Analysis of Thermal Evaluations for Dry Storage Cask of High Burnup SF

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Recently, the US Department of Energy (DOE) has been conducting high burnup fuel demonstration tests through Electric Power Research Institute (EPRI) to extend the licensing of dry storage casks for high burnup spent nuclear fuel (SF) of pressurized water reactor (PWR). EPRI's analysis focuses on optimizing heat transfer and thermal management within dry storage casks. This includes studying the impact of air gap thickness, heat transfer surfaces, and cooling mechanisms on the overall performance and safety of the storage systems. Thermal evaluation of high burnup SF in the dry storage cask are focused on several key such as heat generation and distribution, temperature profiles, cooling systems and thermal stress.

Heat Generation and Distribution: EPRI conducts detailed analyses to understand the heat generation characteristics of high burnup used nuclear fuel and how heat is distributed within storage systems. This includes studying the decay heat production rates and distribution patterns over time.

Temperature Profiles: EPRI evaluates temperature profiles within dry storage casks to assess the thermal performance and safety of storage systems. This involves measuring temperatures at various locations within the casks and analyzing temperature distributions under different operating conditions.

Cooling Systems: EPRI investigates the effectiveness of cooling systems integrated into dry storage casks for high burnup spent nuclear fuel. This includes evaluating the performance of passive and active cooling mechanisms in managing heat dissipation and maintaining safe storage temperatures.

Thermal Stress Analysis: EPRI conducts thermal stress analysis to assess the structural integrity of dry storage casks under thermal loading conditions. This involves evaluating the impact of temperature gradients and thermal expansion on the mechanical behavior of storage systems.

Keywords: Thermal evaluation, Spent fuel, Temperature profile, Thermal stress, Heat generation

Ventilation Control Analysis on SNF Dry Storage Facility Using Fire Modelling Tool

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It has been storing spent fuel (SF) at dry storage facility in Nuclear Power Plants. The aims of SF dry storage facility are to retain safety radioactive level and sustain operation of a nuclear power plant. Most of countries attempt to development a variety of dry storage facility for spent fuel in order to retain the safe storatation. Therefore, many studies have been conducting safety evaluation for the dry storage facility.

However, it does not exist safety evaluation for the ventilation system due to fire event that could influence of the side effect on the dry storage facility since it is likely to occur fire events such as wild fire, air craft crash. In practice, it happened to catastrophic disaster near to ul-jin mountain due to the wild fire. Furthermore, it happened to fire accident near to the Japonia NPP in Ukraine territory dueto military air plane missile in 2023. However, there are not studied on the ventilation safety evaluation considered to fire safety in the SF dry storage facility.

It can be required to the mechanical ventilation systems included in HVAC (Heating Venting and Air Conditioning) system at the SF dry storage system to retain safety conditions versus fire effect. Accordingly, in this study, we conducted to the thermal ventilation control modelling by using fire modelling tool (Fire Dynamic Simulator v.6.7) based on Computational Fluid Dynamics. The ventilation scenarios are consisted of three cases to be difference from the flow rate variation in line with ventilation conditions.

As a result of analysis, it revealed that there would not have differentiation between normal fire ventilation control with proportional smoke ventilation control even though the flow variations and pressure fluctuation were increased.

Second, it evaluated that the control mode for fraction of the flow rate control tends to change the pressure and flow rate in the patterns of rise in the state of controlling the ventilation system flow rate. However, the sensitivity of flow rate will be more decreased by less than 11 seconds because of flow rate control.

Third, it revealed more higher sensitivity resolution than other cases by using flow rate control method caused by kinematic pressure variation.

Fourth, in the case of PID (Proportional Integrated Derivation) will likely to have an influence on improving the smoke control system since it can be adapted to HVAC system possible to controlling smoke layer height, smoke temperature.

These results will be used to the basis data as considering the design guidelines for applying to SF dry storage facility to improve the thermal conditions that can retain fire safety.

Also, it expected that HVAC system installed in dry storage will help to develop automatic ventilation control.

Keywords: Fire dynamic simulator, HVAC ventilation analysis, SF dry storage facility, PID control

Evaluation of Critical Flaw Sizes at Geometric Discontinuities in Spent Nuclear Fuel Storage Canister Using Finite Element Method

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In this study, the fracture evaluation of a spent nuclear fuel storage canister was conducted. For the canisters, defect inspections are required using radiographic or ultrasonic testing after manufacturing. It is essential that the smallest defect detectable in these inspections is smaller than the critical flaw size that is the maximum size of a defect for maintaining structural integrity. Although canisters, typically made of stainless steel, do not require analytical fracture evaluations, this critical flaw size becomes a crucial parameter determining the required accuracy and resolution of the inspection. Therefore, the critical flaw size must be determined through fracture evaluation before performing defect inspections.

The critical flaw size for spent nuclear fuel canisters can be determined using the ASME BPVC Section XI, which provides guidelines for determining the critical flaw size for cylinders and flat plates. However, there is no analytical method provided for determining the critical flaw size at geometric discontinuities, such as the lid-to-cylinder and bottom-to-cylinder welds.

Therefore, this study performed a fracture evaluation based on the finite element method to determine the critical flaw size at the geometric discontinuities of the canister.

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

Evaluation of the Radiological Impacts of Applying ICRP Publication 103 on Spent Fuel Transportation Casks

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Currently, radiological evaluations for spent fuel transportation casks are performed based on ICRP Publication 60. The dose conversion coefficients for photons and neutrons provided in ICRP Publication 74, which are based on ICRP Publication 60, are used for the radiation shielding evaluation of the transportation casks. The A_2 value for the mixture of radionuclides released from the spent fuel into the casks must be calculated for the containment evaluation to determine the allowable leakage rate, and the A_2 value for the IAEA SSR-6 currently in use is also based on ICRP Publication 60. In 2007, the ICRP published ICRP Publication 103, which replaced ICRP Publication 60, and various dose-related coefficients have since been calculated based on it. In the near future, national regulation will also be revised based on ICRP Publication 103, and the IAEA TRANSSC has recently calculated A_2 values based on ICRP Publication 103. Therefore, this study aims to evaluate the impact of applying ICRP Publication 103 on the shielding and containment evaluation of spent fuel transportation casks.

In this study, a hypothetical spent fuel cask is considered and shielding material of the cask is forged steel and boron-containing resin. For photons, the application of ICRP Publication 103 had little impact as there is little difference from the existing dose conversion coefficients, and for neutrons, the dose conversion coefficients are decreased in the low- and high-energy regions, but the impact was not significant for the neutron energy region emitted from spent fuel. For the dose rate at 2 meters under normal conditions, which is typically the most important assessment in the spent fuel transport cask shielding evaluation, the impact was found to be negligible at 0.6%. For the A_2 values used in the containment evaluation, the differences in A_2 values have changed significantly for some nuclides, but the changes in A_2 values for the nuclides that are important for the containment evaluation of spent fuel transport casks are relatively small, with changes in effective A_2 values for spent fuel of 5-10%. This study confirms that the application of ICRP Publication 103 has very little impact on the shielding and containment evaluation of spent fuel transportation casks.

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Keywords: Spent fuel, Transport cask, ICRP publication 103, Shielding evaluation, Containment evaluation

A Study of Technical Standards of Overseas SF Dry Storage Facilities for the Design of SF Dry Storage Facilities in Korea

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Currently, nuclear power plants (NPPs) in Korea are stored spent fuel in wet storage pools. However, the capacity of the spent nuclear fuel (SNF) storage pools at Hanbit, Hanwool and Gori NPPs is expected to be saturated by 2030, so there is an immediate need for a SNF management plan and strategy. Therefore, “High-Level Radioactive Waste R&D Roadmap” and the enact a “High-Level Radioactive Waste Special Act” are being worked by The Korean government. Also, The dry storage system is being promoted project for SNF to efficiently manage PWR SNF generated and stored in NPPs by KHNP. In this study, the SNF dry storage system (DTS, WCS CIFS, etc.) developed by Orano TNA (USA), the design requirements of the Mutsu SNF storage facility designed and constructed by KOBELCO & Orano TN Tokyo (Japan), and the US NRC regulations and Standard Review Procedures (SRP) were analyzed for application to the design of the Korean SNF dry storage facility. The SNF storage system consists of the SNF storage unit, the SNF receiving facility, and the instrumentation and control system facilities, each of which is included in the storage building. The cooling system of the SNF dry storage facility has been designing a passive system, which is designed to adequately remove the decay heat of the SNF assemblies to maintain the integrity of the SNF assemblies and the integrity of the component materials with design basis safety functions. (a) They should be designed to draw air from the surface of the storage casks to the air around the storage casks and to the outside of the SNF storage building using the ventilation force generated by the heat transferred to the air around the casks. (b) to ensure adequate ventilation, the exhaust vents in the storage area and the exhaust vents in the receiving area shall be located at sufficient levels. (c) Temperature detectors shall be installed at the supply and exhaust air vents to measure the supply and exhaust air temperatures of the SNF storage building to monitor that the cooling function is maintained. BRnC Inc. has technology that can be applied in practice through technical cooperation with Orano TN in the field of dry SNF storage systems. We'll be providing technical know-how that will develop an overall management plan for a Korean dry SNF storage system from a technical, safety and economic perspective.

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Keywords: Dry storage systems, Dry storage facilities, Intermediate storage facilities, Spent nuclear fuel management, Radiation protection

Evaluation of Structural Integrity of HIC-510 L Container Based on the Technical Standards of Type a Transport Package

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Safe management of radioactive waste is important to protect humans and the environment. If radioactive waste is released due to physical impact during transportation and storage, the safety of the environment and humans can be threatened. This study evaluates the structural Integrity of HIC-510 L container for transportation based on Type A transportation package requirements. Requirements are 1) There should be no loss or dispersal of radioactive contents. 2) There should be no loss of shielding ability that would increase the radiation dose rate on the outer surface by more than 20% under the normal condition of transportation stipulated in the IAEA Specific Safety Requirements. The safety tests for normal condition of transportation consist of drop tests, penetration tests, stacking test. In addition, the strength of lifting apparatus also needs to be tested as a common requirement for all types of transport packages.

For the drop analysis, the lowest end of the cask was subjected to a free fall from a height of 1.2 meters above the unyielding surface. It is difficult to accurately predict the specific geometry and development of cracks in polymer concrete and the resulting reduction of shielding capability through drop analyses. For the penetration analysis, the HIC-510 L cask was subjected to impact condition where a mild steel rod with a diameter of 3.2 cm and a mass of 6 kg fell onto the cask and lid from a height of 1 meter. The analysis results in some damage and deformation to the concrete and cask lid, but it is not expected to affect the structural integrity assessment criterion. The stacking analysis was performed considering a 7-container stacking situation. The maximum equivalent stresses of the outer shell, inner container, and lid, the maximum principal stress in the concrete were found to be within the allowable stress limits. The lifting analysis was evaluated with 3 times the weight of the transport container applied to the lifting apparatus. There was sufficient safety margin in the external shell, concrete, internal cask made of PE material, and lid.

The study demonstrated that the HIC-510 L container satisfies the technical requirements for a Type A transport package by numerical simulations. However, the detailed geometry of the cracks and the reduction in shielding performance of the concrete were difficult to predict with numerical simulations and physical tests followed by non-destructive testing are required to confirm the safety of the container in terms of radiation shielding.

Keywords: Structural integrity, Finite element mnalysis, Type a transport package, Radioactive waste

Preliminary Study for Transient CFD Simulation of Vacuum Drying

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Predicting the temperature of spent nuclear fuel during dry storage is very important to ensure the integrity of nuclear fuel. The fuel temperature rarely changes during the dry storage phase, so most simulation codes assume the temperature estimation simulation as a steady state. Therefore, the heat capacity term in the energy equation affecting temperature change as a function of time is neglected. However, temperature prediction in the vacuum drying process has been important to estimate because spent nuclear fuels suffer not only the highest temperature but also the fastest temperature change in the vacuum drying process. The simulation should be a transient simulation because temperature change is relatively rapid.

Herein, literature that conducts transient simulation in dry storage systems are reviewed and the CFD transient simulation method is investigated. The heat capacity of fuel assembly for the porous model is addressed. The major heat transfer mechanism in the vacuum drying phase is radiation. A preliminary CFD simulation is presented in this study, neglecting convection contribution. The drying procedure is an important parameter to determine the highest temperature of the fuel assembly.

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Keywords: Dry storage, Vacuum drying, CFD simulation, Transient simulation

International Development Status of Wireless Monitoring Systems for Spent Fuel Dry Storage Facilities

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The global nuclear power industry is increasingly transitioning towards the storage of spent nuclear fuel in dry environments. Effective management of such fuel in dry storage facilities necessitates continuous monitoring of critical parameters including temperature and pressure. Real-time monitoring of the integrity of sealed dry storage containers emerges as a fundamental requirement for ensuring safety. Internationally, efforts are underway to develop non-destructive testing-based monitoring technologies tailored for real-time assessment of dry storage container integrity.

This paper investigated the current status of the development of wireless sensor systems intended for deployment inside stainless steel dry storage canisters.

Keywords: Spent fuel dry storage, Wireless monitoring system

A Study on Material Properties of Concrete in Dry Storage Facilities After Fire Test

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In the case of dry storage facilities, lots of dangerous phenomena are existed that are able to make widespread safety problem. For this reason, in dry storage facilities, evaluations of safety under accidental conditions are important. Accidental conditions in dry storage facilities include natural phenomena such as floods, tornadoes, tsunamis, typhoons, earthquakes, and artificial phenomena such as airplane crashes, fire. However, among phenomena, fire is one of phenomena that is focus on temperature. High temperature is affected on not only radioactive materials but also facilities materials like concrete. The concrete is degraded by high temperature and it lead to loss of facilities strength. So, degraded material properties should be concerned.

In this paper, high temperature analysis was performed based on concrete materials. The reasons of degradation by temperature in concrete were explained using physical mechanism and chemical mechanism. To evaluate concrete degradation material properties, papers were reviewed which were studied about fire test under various temperature situation and summarized degradation rate of degradation concrete compare with normal concrete. The target temperature was 20°C, 100°C, 200°C, 300°C, 400°C, 500°C, 600°C, 700°C and 800°C. Degraded compression stress, degraded elastic modulus, degraded compression strain and degraded tensile stress (split test) were evaluated and formula were suggested to predict degradation material properties at target temperatures. Furthermore, thermal analysis were performed to predict concrete temperature under 800°C temperature, 30 minute, surround heated condition. Concrete wall model was used in thermal analysis and temperature contour was evaluated after thermal analysis. Furthermore, impact analysis was performed to evaluate the safety of concrete wall which was degraded by fire condition.

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Keywords: Concrete storage structure, Fire, Concrete, Degradation, Accident condition evaluation

Review of Technical Requirements for Vacuum Drying Equipment for Nuclear Fuel Sealed Containers

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The Post-Irradiation Examination Facility (PIEF) of Korea Atomic Energy Research Institute operates with the purpose of evaluating the performance of nuclear fuel, developing new fuels, and identifying the causes of damage. Over multiple occasions, a total of 1699 rods of spent nuclear fuel have been transported to the PIEF for various non-destructive and destructive tests. As a result, these fuel rods are stored in the PIEF pool in forms such as assemblies, rods, fragments, and specimens. However, several years ago, the local community continued to criticize the storage of nuclear fuel in urban areas. Because of this, KAERI introduced a sealed container called Quiver from Westinghouse to store and transport nuclear fuel more safely. Considering that these sealed containers ultimately need to be transported to an interim storage facilities, it was necessary to introduce drying equipment to maximize the functionality of the containers. First, since the sealed container is exposed to a wet environment in the pool, equipment with the functions of water evacuation, vacuum drying, and filling helium inside the container is required, and the technical requirements for this were reviewed. The technical requirements for drying equipment were set by reviewing water evacuation using compressed air, vacuum drying, helium filling and sealing inspection, and also PIEF facility application and workability, as well as various technical standards and references. Based on the requirements set in this way, drying equipment was designed, including optimization in PIEF, and is currently in the process of introduction.

Keywords: Drying equipment, Sealed container, Quiver, Post irradiation examination, Damaged fuel

Development of Remote Visual Inspection System for Canister

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The dry storage system at Korea Hydro and Nuclear Power (KHNP) has stored fuel from CANDU reactors since 1992. Engineering data on the condition of the dry storage system is required to ensure long-term safety of the system.

Testing is to be completed on a partial mock-up of a storage canister. The mock-up simulates the top portion of the canister in order to test securing the tool to the canister. The mock-up also contains three empty fuel baskets within a cylindrical liner (compared to nine fuel baskets in the canister) to simulate the geometry the narrow clearances that the tool must navigate. The relative position of the fuel baskets can be adjusted in the mock-up.

The temperature sensor and the radiation sensor will not be tested in the mock-up for the extreme environments for which they are intended (100°C and 5,700 Rad/hr). The RTD has a specified measurement of up to 260°C and has been calibrated for up to 100°C. The radiation sensor uses a unique electrometer and been calibrated for the range of measurements to be made in the canisters. Therefore, further mock-up testing of the sensors is not warranted because the calibration process demonstrates functionality of the sensors. When the sensors are installed on the RVI tool and the data acquisition system, a simple reading of the ambient conditions will demonstrate the sensors are functional in the RVI system.

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Keywords: CANDU spent fuel, Silo, Canister, Basket, Remote visual inspection system, Mock-up

First Interim Storage Examination of Bundles for CSF

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The first ISE (Interim Storage Examination) occurred after approximately 41 months of storage in air at 150°C. The ISE was performed at the hot cell facility at WL (Whiteshell Lab.) In Canada, and included a visual examination of the storage container, preliminary examination of the fuel bundles, and a detailed examination of specific fuel elements removed from the bundles. Along with control element 1, outer elements 2 and 3 were examined during the first ISE. The storage container and bundles were removed from the basket in the WL hot cell facility. All containers were covered with black iron oxide, and the containers with defected bundles were also coated with various rust-coloured deposit. SEM/EDX analysis indicated only iron. All fuel bundles appeared unchanged from their pre-storage condition.

During ISE-1, since control element 1 was intact, intentionally-defected element 2 was characterized, and then designated as the “defected control element”; therefore, further destructive examinations were not conducted on element 2. This allowed for the storage behavior of both an intact and defected control element to be monitored and compared in future examinations.

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Keywords: CSF, ISE, SEM/EDX analysis, Intact

Analysis of Test Result for Metallographic Examination of ISE

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Sections of cladding from various positions along the length of intentionally-defected element 8 were examined to determine if the fuel sheath had experienced any degradation during the storage period. A silicon carbide blade cooled with an argon cover gas (to prevent any oxidation of the sheath) was used to cut all the samples. The inner and outer sheath surfaces were examined for incipient cracks, and the Zr H/D distribution was examined in each of the sections to determine if any H pickup or H/D migration had occurred during storage.

The Zr oxide layer on the outer sheath surface and the CANLUB layer on the inner surface were also examined to determine the effect of storage on them. The bearing-pad/braze alloy and the sheath/endcap weld regions were also examined to determine if any deterioration had occurred during storage. There was no significant difference between the results of the metallographic examination performed on the sheath before and after the storage period. No incipient cracks were observed on either the inner or outer sheath surfaces. The oxide layer on the outer sheath surface varied from 2 to 3 μm in thickness before and after storage. A buildup of zirconium oxide was observed on the beryllium/zirconium braze alloy in the sheath/BP braze regions. The CANLUB was observed adhering primarily to the sheath.

There appeared to be no significant difference in the Zr H/D distribution in the cladding before storage, and after the storage period.

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Keywords: CSF (CANDU Spent Fuel), Metallographic examination, ISE, Cladding

Development of the Evaluation Program Module for the Storage of SF

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Two evaluation modules were developed to evaluate the suitability of various linked scenarios considering storage in nuclear power plants and to develop a data system for the management of spent fuel linkage.

First, KHNP developed a radioactivity evaluation module when transporting SF. The spent fuel dry storage automation module can calculate the heat of decay and the amount of radiation immediately after the spent nuclear fuel is released and select the spent nuclear fuel according to the number of stores. In addition, the location can be selected according to the spent nuclear fuel storage mechanism, and the output result can be provided by calculating the stored fuel in detail.

Second, due to the change in the domestic SF characteristic data production code system, it is necessary to manage the data base of characteristic data and develop a linked data production system. In order to build the ASTRA code system result, which is a core design code, into a data base and increase the ease of user interpretation, developed a methodology that can be visualized and linked to related systems.

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Keywords: Radioactivity evaluation, Characteristic data, Spent fuel, Evaluation module

Test Result of Dry Storage Inspection System for Canister (Silo Mock-Up)

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A dry storage inspection system (Remote Visual Inspection: RVI) was developed to facilitate the visual inspection of the inside of dry storage canisters for CANDU Spent fuel, as well as the fuel baskets within it. To confirm the performance of the developed RVI, a test was performed using a mock-up according to a verifiable plan.

The mock-up area is located at the base of the tower in test factory., where there is sufficient headroom and a 5-ton crane for use. The limited physical space at the base of the tower, coupled with large size of the mock-up and the RVI tool, resulted in numerous challenges during tool assembly, mock-up assembly, and placement of the RVI tool on top of the mock-up. These challenges are noted in the test results to distinguish them from the field conditions, where such physical constraints would not be so pronounced. The tests were performed according to Test Plan to demonstrate the tool satisfies the acceptance criteria, fulfills the design requirements, and is suitable for use in the field.

The performance test includes visual inspection of the camera, temperature and radiation dose measurement, which are the main inspection items. It also checks the smoothness of the inspection by passing about an inch between the silo inner liner and the basket, and the bottom of the canister by passing through the inside of the salvage rod. In the future, if it is decided that an internal inspection of the actual canister is necessary, a plan will be established for that.

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Keywords: Candu spent fuel, Silo, Storage canister, Basket, Inspection system, Mock-up

Requirements of Long-Term Dry Storage Experiment for CEX

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Due to the delay in securing the disposal site, the final management plan for spent nuclear fuel, and the recycling technology is not developed smoothly in most countries, dry storage of spent nuclear fuel by power generators is becoming prolonged worldwide. With the decades-long licensing period, each country is preparing test data to serve as the basis for safety evaluation to extend the operating period of the existing spent nuclear fuel dry storage system, and the spent nuclear fuel demonstration program is an inevitable process to ensure such long-term safety.

Canada has been operating three types of CANDU dry storage (Silo/Canister) demonstration systems for 30 years, and this demonstration also checks the soundness of spent nuclear fuel and analyzes gas components through hot cell tests. CEX (Controlled Environment Experiment) was designed to evaluate the integrity of bundles in dry storage conditions in Canada.

As in the case of developed countries, the requirements for demonstration of spent fuel in CANDU reactors can be summarized as confirming the soundness of spent fuel, gas analysis inside the demonstration system, and pressure measurement.

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Keywords: CANDU spent fuel, Long-term storage, Dry storage silo, Demonstration

Review of Dry Storage Application for the Development of SF Management Data System

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The integration of a data system for managing spent fuel (SF) necessitates a thorough technical analysis of applying dry storage systems. This paper provides a comprehensive examination of the technological aspects involved in utilizing dry storage systems to develop an integrated management data system for SF. By analyzing technical components such as design considerations, structural integrity, radiation shielding, thermal management, and environmental impact, this study aims to elucidate the critical factors influencing the application of dry storage systems in SF management. The findings of this review will contribute to the direction of system regarding the development and implementation of an effective data system for integrated SF management.

The objective is to review the technical standards for the application of dry storage systems domestically and internationally, aiming to identify appropriate technical criteria

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Keywords: Dry storage, Management data system, Design consideration

Review for Conceptual Scenario in Power Plant and Intermediate Storage of SF

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Encapsulation of spent fuel refers to the transfer of the fuel from the former container, where it was stored, to a latter container for future storage. Spent fuel, characterized by high radiation and high heat, poses challenges related to the potential diffusion of fission product gases in the event of partial cladding failure. Therefore, repackaging of spent nuclear fuel must be conducted in specialized facilities such as hot cells, rather than conventional buildings or facilities, due to safety concerns.

However, nuclear power plants requiring dry storage for spent fuel may face saturation of wet storage capacity within the reactor pool over extended operational periods, or necessitate dry storage means during decommissioning processes to remove spent fuel before reactor shutdown. Consequently, the construction of encapsulation (repackaging) facilities on or off-site presents challenges and safety concerns, hence minimizing the frequency of repackaging operations is crucial. Therefore, when considering the scenario, it should be considered that the encapsulation can be minimized.

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Keywords: Candu spent fuel, Silo, Storage canister, Basket, Inspection system, Mock-up

Review of DOE's Work on the Disposition of SF in Dual-Purpose Canisters

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In the United States, commercial SF is stored at more than 70 sites and continues to be generated at a rate of more than 2,200 metric tons of heavy metal per year. Much of the commercial SF inventory has been stored inside dry-storage casks at independent spent fuel storage installations (ISFSIs) associated with nuclear power plants because the spent fuel pools at the plants do not have the storage capacity to accommodate all the SNF discharged from the reactors.

In the absence of a clearly defined disposition pathway for SF, long-term dry storage has become the SF management program in the United States for the foreseeable future. Much of this inventory is in dry storage inside dry-storage casks, most of which are large, welded canisters referred to as DPCs that have been designed for interim storage and transportation but not for their potential use for geologic disposal.

The three principal alternative approaches for managing SF in an integrated waste management system are (1) indefinite storage (for more than 80–120 years) at ISFSIs, (2) repackaging the SF into smaller canisters before disposal, and (3) direct disposal of SNF in DPCs in a geologic repository. Each of these alternative approaches has implications for the SF management and disposal system. DOE has examined, in a variety of past evaluations, the pros and cons of each alternative approach and has developed a number of useful analysis tools well-suited for these types of evaluations.

DOE R&D activities related to direct disposal of SF in DPCs currently focus on three areas: (1) the consequences of potential criticality events on the long-term performance of a geologic repository after closure; (2) potential filler materials that could be injected as liquids into existing DPCs, where they solidify and prevent groundwater ingress into breached DPCs and, thereby, reduce the probability for nuclear criticality; and (3) modifications to future DPCs so they will remain subcritical in any repository setting.

Keywords: DOE, DPC (dual purpose canister), Spent fuel, Long-term dry storage

Review of DOE's Plans to Test Full-Size Spent Fuel Transportation Cask

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The US Department of Energy (DOE) is planning to test a full-size spent fuel transportation cask in an operation known as a Package Performance Demonstration (PPD). Department of Energy provided details of the planned demonstration at the Institute of Nuclear Materials Management's recent Spent Fuel

This demonstration is intended to accomplish several objectives, including: First is to build trust and confidence in the safety of spent fuel transportation casks and spent fuel transportation by rail. Second is to gather technical data to further validate computer models. Third is to record high-resolution video to use in DOE communication products and public outreach. Fourth is to consider public engagement and stakeholder outreach strategies in support of the PPD. Last thing is to explore additional opportunities to strengthen relationships between DOE and the public, further enhancing DOE's efforts to build trust.

A preliminary Functions and Requirements Document was completed in September 2023. It is a multi-laboratory effort that includes information such as functions and requirements for potential tests to be performed, test sites, transportation casks, and analytical tools. Stakeholder support is essential to a successful demonstration program. DOE will lead the PPD team, which includes experts from PNNL, Sandia National Laboratories (SNL), Oak Ridge National Laboratory (ORNL), and Savannah River National Laboratories (SRNL). DOE will continue external engagement regarding the PPD, which is dependent on availability of funding to complete the program.

Keywords: DOE, Transportation CASK, PPD, Spent fuel, Demonstration

3분과

고준위폐기물처분 (ORAL)



Assessing Radiation-Induced Corrosion Damage on Copper Canisters in KAERI Reference Disposal System Using Numerical Simulation

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Radiation-induced corrosion of the copper (Cu) canister is one of the major concerns for the licensing of a deep geological repository. There have been a lot of efforts made internationally to evaluate the damage of radiation-induced corrosion of Cu canisters, notably by SKB, Posiva Oy, and NWMO. In despite of these efforts, the estimated corrosion damage for the KAERI Reference disposal System has not been reported yet. Hence, the authors have developed a Multiphysics model to evaluate the corrosion of Cu canisters via gamma-ray induced reactions for the KAERI Reference disposal System (KRS). The developed model is based on the Multiphysics Object-Oriented Simulation Environment (MOOSE). In the developed model, the radiolysis of groundwater was considered by 37 dominant chemical reactions for water with kinetic constants. PLUS-7 type spent nuclear fuels with a $45,000 \text{ MWD} \cdot \text{MtU}^{-1}$ were considered to calculate the absorbed dose rate in the near-field. Also, the temperature distribution in the near-field was simulated with thermal-hydraulics governing equations. The maximum corrosion depth is calculated to be $0.23 \text{ } \mu\text{m}$ after a period of 10 million years. This calculation does not take into account the trapped O_2 in the near-field of the KRS. Also, the simulation result indicates that O_2 is not the only major oxidizing agent, but Cu^{2+} is also found to be a major oxidizing agent. The predicted corrosion potential is $-0.4 \text{ V}_{\text{SCE}}$ to $-0.7 \text{ V}_{\text{SCE}}$. This value is considerably lower than the pitting corrosion potentials of 0.2 to 1 V_{SCE} for 25 and 90°C Cl^- containing solutions. These results indicate that the predicted corrosion damage is very small even after 10 million years. However, the results of this simulation are a primary study for KRS, therefore additional research is required to evaluate the radiation-induced corrosion of Cu canisters in KRS.

Keywords: Radiation-induced corrosion, Copper, Canister, Gamma-ray, Radiolysis, Multiphysics

Quantification of Risk Profile for Deep Geological Disposal Repository: A Comparison With TSPA

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According to the notice of Nuclear Safety and Security Commission (NSSC), the main safety objective of risk assessment for deep geological repository is designated by risk and it should be presented in the form of a profile. The methodology of Total System Performance Assessment (TSPA), which has been widely used for safety assessment of disposal facilities, is suitable for assessing dose criteria but faces challenges in deriving the risk level that integrates all scenarios. Moreover, the risk assessment methodologies based on event trees and fault trees, generally applied to nuclear power plants, are difficult to utilize directly due to the unique characteristics of disposal facilities. Against this backdrop, various studies are being conducted to appropriately evaluate the risk of deep geological repository.

In this paper, we propose a method to quantify the risk of integrated scenarios, employing the GoldSim code, which is broadly used for nuclide transport assessment. With this tool, we simulated similar conditions to those of a domestic deep geological repository for spent nuclear fuel, where both normal and abnormal scenarios coexist. Through a case study using this model, the paper elaborately discusses the proposed risk quantification approach. This risk quantification procedure is expected to aid in understanding the risk assessment requirements demanded by domestic notice.

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Keywords: Deep geological repository, Spent nuclear fuel, Risk assessment, Risk quantification, Total system performance assessment

Structural Investigation of Alkali Aluminoborosilicate Glass Containing MoO₃ for Vitrification of Nuclear Waste

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Molybdenum (Mo), a prevalent fission product in high-level radioactive waste, presents challenges due to its low solubility and tendency to undergo undesirable crystallization in nuclear waste glasses. Increased concentration of MoO₃ promotes the formation of crystalline phases (e.g., Na₂MoO₄, CaMoO₄) in the glass network, potentially harming the long-term property of nuclear waste glasses. In this study, the effects of incorporating various alkali cations (Li, Na, K, Cs) into Mo-containing aluminoborosilicate glasses were investigated using XRD, SEM, DSC, XAS, XPS, Raman, and solid-state MAS NMR spectroscopy. When taken together, the results indicate that incorporating larger alkali cations leads to a more polymerized silicate network, which in turn hinders the crystallization kinetics. The spectroscopic results suggest that the structure of Mo units changes with the increasing size of alkali cation, which could be manifested in two possible ways. Both possibilities enable direct connection of Mo units with the silicate network (Mo-O-Si) considering Pauling's stability rule. The formation of the Mo-O-Si bond attributes not only enhances the solubility of Mo in aluminoborosilicate glasses, but also prevents the crystallization of Mo-containing phase by inhibiting the formation of nucleation seed, MoO₄²⁻ entities.

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Keywords: Molybdenum, Aluminoborosilicate glass, Glass structure, Nuclear waste vitrification

Radioactive Waste Management in Germany

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Radioactive waste management in Germany is a state-owned task. The responsibility within the Federal Government lies at the Federal Ministry for the Environment, Nature Conservation, Nuclear Safety and Consumer Protection (German abbreviation BMUV). As regulator, the BASE, the Federal Office for the Safety of Nuclear Waste Management, is implemented. With regard to legacy waste, the tasks of the regulator lie partly within the different states. For the operation of the different waste management projects, BGE (Federal Company for Radioactive Waste Disposal) was established. BGE is responsible for the construction and eventual operation of the LILW repository Konrad, the operation and retrieval of waste from the Asse II mine, the operation and eventual closure of the Morsleben site, the closure of the former exploration mine Gorleben as well as the site selection process for a future HLW/SNF repository in Germany. The site selection procedure was restarted in 2017. Currently, potential site areas are being screened. As Germany has a diverse geology, the site selection process includes three different host rock formations: salt, claystone, and crystalline rock. The existing facilities Gorleben, Asse, and Morsleben are located in salt formations. The emplacement level of the Konrad site is located in a limestone/iron ore formation, covered by a massive clay formation. In addition to the construction and operation business, BGE holds the subsidiary BGE TECHNOLOGY GmbH, which carries out R&D for a future HLW/SNF repository, maintains and intensifies international cooperation, and transfers knowledge into BGE. To fulfil these tasks, BGE TECHNOLOGY GmbH is involved in numerous projects and collaborations.

Effect of Different Copper Loadings for Efficient Iodine Capture on Metallic Copper Decorated Hexagonal Boron Nitride Under Dry and Humid Streams

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Radioiodine isotopes, particularly ^{131}I , in the form of molecular iodine (I_2), pose significant environmental concerns due to their high volatility and the potential release during the reprocessing of spent fuel or during nuclear accidents. The need to address these concerns and to develop new materials with efficient iodine capture and safe storage capabilities in off-gas streams is crucial for mitigating environmental release and health risks.

Among the reported iodine sorption materials, copper-based materials have shown promise for the radioactive iodine capture; however, consideration for the impact of water vapor was deemed essential within a real reprocessing off-gas stream, as its presence often hampers iodine capturing due to the competitive sorption. Herein, metallic copper decorated hexagonal boron nitride ($\text{Cu}^0/\text{h-BN}$) composites were synthesized via a straightforward solvothermal reduction route, incorporating four different copper (Cu) loadings with the aim to assess their sorption performance under different conditions (dry and humid conditions). Hexagonal boron nitride (h-BN) was chosen as an ideal substrate to provide a stable platform for copper decoration and enhancing the overall sorption performance.

The exploration of different Cu loadings (25, 56, 80 and 89wt%) aimed to identify optimal Cu loading for iodine capture. Through iodine capture experiments, it was observed that $\text{Cu}^0_{89\text{wt\%}}/\text{h-BN}$ showed little degradation under highly harsh conditions, reaching a maximum iodine capacity of $1,876 \pm 119$ mg/g in air, $1,843 \pm 116$ under 18% relative humidity, and $1,841 \pm 123$ under 75% relative humidity. The results highlight the substantial influence of Cu loading on enhancing the iodine sorption capacity of $\text{Cu}^0/\text{h-BN}$ composites.

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Keywords: Radioactive iodine capture, $\text{Cu}^0/\text{h-BN}$ composites, Cu loadings, Relative humidity

Impact of Organic Acids on the Dissolution of Iron-Uranium Simulated Fuel Debris

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At the Fukushima Daiichi Nuclear Power Station (1F), the molten fuel reacted with Zircaloy cladding at high temperatures, forming fuel debris. Particularly, when the air entered the reactor due to a hydrogen explosion, UO_2 would be oxidized and reacted with Fe in the reactor structure materials, forming the $\text{Fe}^{\text{III}}\text{U}^{\text{VI}}\text{O}_4$ phase. Fuel debris removal at 1F is underway, and the debris will be disposed of deep underground. Understanding the dissolution behavior of fuel debris in groundwater is essential for the long-term safety assessment of waste disposal based on migration scenarios. Groundwater generally contains natural organic matter, NOM. Among the functional groups in NOM, a carboxyl group mainly forms a metal complex with radionuclides from fuel debris. The organic ligands also possibly increase the dissolution of fuel debris, but few studies have been conducted.

The present study evaluated the dissolution behavior of FeUO_4 in the presence of dicarboxylic ligand (malonic acid, $\text{CH}_2(\text{COOH})_2$) as a relatively simple structure of NOM. The congruent dissolution of Fe and U of FeUO_4 was observed in the early stage of the sample immersion. A higher concentration of malonic acid increased the dissolution rate of FeUO_4 . Under a steady state condition after the long-time immersion, Fe and U valence states were trivalent and hexavalent in the solution, respectively, similar to the previous study in the absence of malonic acid.

Thermodynamic considerations suggested that the measured concentrations of trivalent Fe and hexavalent U in the neutral pH region were controlled by assuming ferrihydrite ($\text{Fe}(\text{OH})_3$) and metaschoepite ($\text{UO}_3 \cdot 2\text{H}_2\text{O}$) as the solubility-limiting solid phases, respectively. The predominant aqueous species were $\text{Fe}((\text{OOC})\text{CH}_2)_3^{3-}$ and $\text{UO}_2((\text{OOC})\text{CH}_2)_2^{2-}$. Meanwhile, the solid phases at weakly acidic pH would be Fe-malonate and U-malonate. In summary, malonic acid increased the apparent solubility of Fe and U due to the formation of soluble organometallic complexes and could also form some malonate-containing solid phases at a given pH, depending on the malonic acid concentration.

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Keywords: FeUO_4 , Organic acid, Fuel debris dissolution, Fukushima daiichi nuclear power station

Adsorption Behavior of Iodate (IO_3^-) and Iodide (I^-) on Layered Double Hydroxides

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Iodine, a long-lived fission product, exists in multiple physical states such as solid, liquid, and gas and multiple oxidation states (-1, 0, +1, +5, +7). The source of aqueous iodine is primarily from reprocessing of spent fuels. At the Hanford sites, iodate (IO_3^-) constituted 70% of the total iodine, with the remaining iodide (I^-) and organo-iodine. Similarly, the Savannah River site reports showed that 80% of the total iodine consisted of iodate (IO_3^-), and the remaining were iodide (I^-) and organo-iodine. These results were in contrast with thermodynamic prediction due to the presence of sedimentary organic matter and the relatively high pH of sites. Moreover, an attentive decontamination method for iodine is important due to its high mobility for long-term disposal.

The iodide (I^-) and iodate (IO_3^-) have different chemical characteristics; iodide (I^-) is regarded as a soft base, whereas iodate (IO_3^-) is a hard base. Thus, the tendency of functional species such as Cu, Ag, and Bi to capture iodine was different depending on the iodine species. Conventional silver-based adsorbents such as Ag-zeolite and Ag-activated carbon effectively capture the iodide (I^-) but cannot capture the iodate (IO_3^-). Thus, the development of new adsorbents for iodate (IO_3^-) is necessary.

Layered double hydroxides (LDHs) are one of the promising adsorbents for iodate (IO_3^-) due to the various adsorption mechanisms such as anion layer exchange, memory effect, and surface adsorption. Among these mechanisms, surface adsorption as a function of metal compositions is a promising research area in this field due to its compositional flexibility. Thus, in this study, we have set the systematic study for interpreting the adsorption behavior of both iodate (IO_3^-) and iodide (I^-) on the LDHs as a function of metal types such as Mg, Co, Ni, Cr, and Al with the integrated characterization.

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Keywords: Layered double hydroxides, Iodate, Iodide, Adsorption, Decontamination

Assessment of Gas Generation and Transport in Geosphere: Implications for Deep Geological Repository Safety

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Deep geological repository for radioactive waste disposal is designed with multiple barriers to prevent the leakage of radionuclides. Among the various processes affecting repository safety, gas generation and transport in the geosphere present significant challenges. Methane generation, in particular, is a critical concern due to its potential impact on the mechanical and chemical stability of the repository system. In the geosphere, gas generation arises from the microbial degradation of organic matter and the radiolysis of water. These processes can lead to the accumulation of gases, including hydrogen and methane, which may impact the integrity of engineered and natural barriers. Methane, being less soluble in water compared to other gases, can accumulate and form gas phases under certain conditions, influencing the pressure and flow of groundwater around the repository.

The behavior of gases in the geosphere is complex and influenced by the geological structure, the presence of fractures, and the hydrogeological conditions. Gases can migrate through porous rock and fracture networks, potentially leading to their release into the biosphere or affecting the transport of radionuclides. Understanding the interaction between gas generation, migration, and containment within the geosphere is crucial for assessing the long-term safety of deep geological repository. Research and experimental studies are focused on quantifying the rates of gas generation, investigating the mechanisms of gas transport, and assessing the impact of gas on the repository's safety. These studies include modeling the behavior of gases under various scenarios, including changes in temperature, pressure, and geochemical conditions.

Ensuring the containment and isolation of radioactive waste requires a comprehensive understanding of gas-related phenomena within the geosphere. Continuous research and monitoring are essential to mitigate the risks associated with gas generation and transport, thereby enhancing the safety and reliability of deep geological repository for radioactive waste disposal.

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Keywords: Nuclide behavior, Gas generation and transport, Geochemical circulation, Geosphere, Deep geological repository

3분과

고준위폐기물처분 (POSTER)



IAEA's Roadmap for Implementing a Geological Disposal Programme: Facilitating Knowledge Transfer Between Advanced and Early-Stage Programmes

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The IAEA has issued a technical publication 'Roadmap for Implementing a Geological Disposal Programme'. The roadmap identifies the major activities and deliverables for each phase of the programme that are the responsibility of the implementing organization to construct and operate a geological disposal repository in a manner consistent with existing international experience. The significant contribution of underground research facilities to this field is highlighted.

The roadmap is generic in nature, describing only the activities common to each phase but not specifying the method to complete the activity or time allotment. Rather, it is presented to reflect good practice and lessons learned, thereby aiming to help reduce risks for new programmes. The publication also captures knowledge (case studies) of how several advanced programmes have progressed along their respective roadmaps toward siting, constructing, and operation of a geological repository. This document forms the basis of a training course that is given to countries embarking on a nuclear programme and those at earlier stages in the development of a deep geological repository programme.

Keywords: Geological disposal, Spent fuel, High level waste, Roadmap, IAEA

Current Status of the Radioactive Waste Management Program in Switzerland - Site Selection Process for a Deep Geological Repository

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The Swiss Nuclear Energy Act requires that all types of radioactive waste must be disposed of in deep geological repositories (DGR). In 1972, the Swiss federal government and the nuclear power plant (NPP) operators founded the “National Cooperative for the Disposal of Radioactive Waste” (Nagra), which is responsible for implementing the permanent safe disposal of all types of radioactive waste generated in Switzerland. According to the 2004 Nuclear Energy Ordinance, the site selection process for a DGR for radioactive waste is defined in a so-called Sectoral Plan and within the framework of spatial planning legislation.

Initiated in 2008 by the Federal Council, a three-stage process, led by the Swiss Federal Office of Energy (SFOE) with Nagra as the implementer, aimed to identify suitable repository sites. This process began with a “white map” of Switzerland, resulting in the selection of siting regions according to safety criteria proposed by Nagra. Approved by the federal government in 2011, six potential siting regions were proposed, three suitable for high-level waste (HLW) in addition to low- and intermediate-level waste (L/ILW).

In the second stage, areas for surface facilities were identified, and the number of siting regions was narrowed down. Following public consultation, three geological siting regions (Jura Ost, Nördlich Lägern, Zürich Nordost) were selected for detailed investigation in Stage 3.

The third stage of the site selection process led to the communication of the proposed site for a combined repository in September 2022 and will result in the preparation of the corresponding general license applications for submission in November 2024.

In addition to the investigations carried out in Northern Switzerland since the 1980s, 3D seismic campaigns covering a total of more than 200 km² were carried out from 2015 to 2017 in the three siting regions. Additionally, an extensive exploratory drilling campaign was conducted in the three siting regions from spring 2019 to spring 2022.

After general licenses are granted, construction preparations will commence, with operations expected to begin around 2050 for low- and intermediate-level waste and around 2060 for high-level waste.

Throughout this process, Nagra collaborates closely with international waste management organizations, universities, and research institutes to ensure the safe construction, operation, and closure of the repository, emphasizing the importance of sharing information and learning from each other.

Keywords: Switzerland, Radioactive waste management, DGR, Site selection

Electrochemical Corrosion Behavior of Copper Fabricated by Additive Manufacturing for Deep Geological Disposal

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Copper canisters are chosen as integral components of engineered barrier systems for the deep geological disposal (DGD) of spent fuel (SF). Copper exhibits a corrosion rate of less than a few millimeters over a million years of DGD and demonstrates uniform corrosion behavior, as opposed to localized corrosion, which poses the risk of releasing radioactive. Due to its thermodynamic stability under anoxic conditions and relatively low cost, copper has gained acceptance as a corrosion-resistant material for SF canister in DGD environments. Copper canisters are traditionally manufactured through extrusion and forging processes in Sweden and Finland. These well-established methods have been rigorously evaluated to ensure compliance with the safety requirements of regulatory authorities. However, the thickness of the copper layer in conventionally manufactured canisters is considerably thicker than that actually required. As the conventional method results in high production costs, alternative techniques aimed at reducing the thickness of copper canisters have been explored. In Canada, the Nuclear Waste Management Organization (NWMO) has adopted copper coatings onto steel substrates via electrodeposition and gas dynamic cold spray, successfully fabricating 3-mm thick Cu canisters with sufficient mechanical strength and corrosion resistance. In South Korea, the Korea Atomic Energy Research Institute (KAERI) has fabricated a millimeter-scale copper film using the cold spray coating technique. Recently, KAERI also have been investigated the technical viability of various additive manufacturing (AM) techniques to fabricate a millimeter-scale Cu-coated film for DGD canister application. In this study, we investigated the electrochemical corrosion behavior of copper fabricated by AM as a canister material in oxic condition. Naturally aerated groundwater from KAERI underground research tunnel (KURT) was used as an electrolyte for the experiments. Potentiodynamic and electrochemical impedance spectroscopy measurements were carried out to measure the corrosion potential, corrosion current density, and corrosion resistance in oxic KURT groundwater solution.

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Keywords: Copper, Corrosion, Canister, Deep geological disposal, Electrochemistry

Feasibility Analysis of Rod Consolidation for Spent Nuclear Fuel Disposal

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Following a consensus on the disposal of High-Level Waste (HLW) in Deep Geological Repository (DGR), numerous conflicts have arisen between nuclear engineers and the public regarding the recognition of Spent Nuclear Fuel (SNF). Various solutions have been proposed to alleviate public concerns, including reducing the repository area through methods such as rod consolidation, despite the need to revise current thermal criteria. In this study, the potential for rod consolidation, which involves combining multiple assemblies into a single assembly, has been examined regarding Thermal-Hydraulic (TH) behavior of Engineered Barrier System (EBS) and its size reducing characteristics. TH analysis was conducted on various types of past, current, and predicted SNF, having various burn-up (30~70 GWD/MTU) and uranium concentration (3~6w/t%) in the KAERI Reference disposal System (KRS). Results show that rod consolidation is feasible for past and current SNF types with low burn-up. However, future SNF types with higher burn-up pose challenges, with temperatures exceeding 200°C. These trends persist across all uranium concentrations. Therefore, while rod consolidation may be viable for certain SNF types, especially pre-current SNF types, it may not be suitable for others, future SNF having high burn-up, necessitating careful consideration and further investigation into SNF disposal strategies.

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Keywords: Effective disposal, Rod consolidation, Thermal-hydraulic

Alteration of Na Bentonite in YS03 Borehole

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The YS03 exploration borehole, situated near the KAERI fence at a depth of 350 m, maintains an anaerobic environment over extended periods. It was characterized by a high concentration of anaerobic bacteria and hydrogen sulfide in the groundwater. Long-term corrosion tests have been ongoing since 2019 using the YS03 borehole. Recently, MX-80, an uncompressed Na-type bentonite, was recovered and subjected to ion analysis, XRD, and XRF. Originally bright blue-gray, the MX80 exhibited red discoloration. Cation analysis revealed a significant depletion of Na cation and an increase in Fe cation due to the corrosion product of cast iron. Additionally, the influence of sulfate-reducing bacteria (SRB) led to the disappearance of most sulfates in MX-80. XRD analysis showed the complete disappearance of Montmorillonite, replaced by Muscovite as the main mineral. Furthermore, significant amounts of Goethite and Magnetite were newly formed due to the influence of Fe cation. XRF oxide analysis confirmed a significant reduction in alumina (Al_2O_3), silica (SiO_2), and magnesium oxide (MgO) components in MX80. In conclusion, it was observed that the alteration of MX80 in the SRB-rich YS03 exploration borehole could proceed much faster than anticipated.

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Keywords: Anaerobic corrosion, Deep geologic disposal, Bentonite, Montmorillonite, Alteration, MX80

Preliminary Study on the Friction Stir Welds Characteristics of Oxygen-Free Copper for Spent Nuclear Fuel Disposal Canister

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Outstanding corrosion resistance of outer copper shell formed for the corrosion protection of disposal canister is crucial for ensuring the long-term integrity of spent nuclear fuel disposal canister. During the sealing process after loading spent nuclear fuel, however, corrosion-sensitive welds formed inevitably and applicable welding methods should be carefully selected considering their corrosion resistances. In this study, the applicability of friction stir welding (FSW) for oxygen-free copper (OFC) of outer disposal canister was experimentally evaluated among various welding processes. FSWs samples were prepared using a specially designed FSW tool 40 mm thick specimen to secure weld joints, and microstructural analysis was performed in welds zone and base material. As a result, nugget and thermo-mechanically affected zone (TMAZ) were not observed, which are formed conventional aluminum materials. The formation of stir zone (SZ) with microstructural refinement due to dynamic recrystallization was clearly observed, resulting in approximately a tenfold decrease compared to the base material. The Vickers hardness results of SZ and heat-affected zone (HAZ) showed no significant increase when compared to the base material. This result indicate that the manufacturing history of OFC was more dominant for hardness changes than the grain size effect. However, incomplete recrystallization in the HAZ showed the fastest corrosion rate, indicating that welding process development and post-weld heat treatment should be considered for improving the corrosion-resistance of the HAZ during FSW process.

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Keywords: Friction stir welding, Oxygen-free copper, Disposal canister, Copper welds, Deep geological disposal

Manufacturing of Engineering-Scale PWR Disposal Canister

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

The inner cast iron canisters of spent nuclear fuel disposal canisters manufactured as a result of the study are combined with outer copper canister technology and are directly used for demonstration of engineering barrier performance in underground facilities (URL) essential for final disposal of spent nuclear fuel. In the last study, CANDU disposal canister was manufactured, and the cold spray method was applied to the outer copper canister. In this study, engineering-scale PWR disposal canister was manufactured, and the flow forming method was applied to the outer copper canister. Flow forming is a hybrid method between forging and casting, and the length of the outer copper canister can be increased by placing a preform forging on the mandrel and using a hydraulic tailstock. The outer copper canister was carried out in the order of copper ingot cutting, preform hot forging, machining and finally flow forming.

Keywords: Disposal canister, PWR, Spent nuclear fuel, CANDU

Varying Disposal Depths in Deep Borehole Disposal of Spent Fuels

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Deep borehole disposal (DBD) of spent nuclear fuels (SFs) has been investigating as an alternative option. DBD can give advantages over less deep geological disposal since the disposal of wastes at a great depth where a low degree of permeability in the potentially steady rock condition will be beneficial for nuclide movement. The DBD concept is quite straightforward and even simple: Waste canisters are simply emplaced in the lower part of the borehole down to more or less than 5 km deep.

Through this study, a conceptual DBD is assessed in view of its sensitivity of the disposal depths to the radiological safety, in which PWR SF canisters (the only high-level wastes in Korea) are to be emplaced at a deep bottom, upon which an additional e.g., compacted bentonite is overbuffered, and the remaining upper part of the borehole is backfilled again with a mixture of e.g., crushed rock and bentonite. Then, the total length of the deep borehole distinctly has three zones: a disposal zone where SFs are disposed of at the bottom, a buffer zone at the next middle layer, and backfill zone at the rest top layer.

To demonstrate the feasibility in view of long-term radiological safety, a rough model for a safety assessment of this conceptual deep borehole repository system, providing detailed models for nuclide transport in and around the geosphere and biosphere under normal nuclide release scenarios that can occur after a closure of the repository, has been developed using GoldSim here at KAERI.

Since an upward groundwater flow along the borehole has not been considered to be that great except for along the EDZ, only diffusion transport through the bentonite hollow between the waste canister and the inner surface of the borehole, as well as the upper buffer layer in the buffer zone is assumed to be the dominant transport mechanism. Groundwater flow and advection transport are assumed to occur only through the EDZ, to which all the nuclides released from the disposal zone and buffer zone are collected, and from which nuclide transfer to the fractured rock medium surrounding the borehole takes place for farther transport to the MWCs, which might provide the shortest and fastest ways for nuclides into the biosphere.

Nuclides from the backfill zone and EDZ at the uppermost part of the borehole meet an aquifer where the groundwater flows toward the surface water bodies in the biosphere, as depicted in the figure.

Since some trade-offs are driven by economical aspect between the disposal depth and the assurance of radiological safety it could be worth to investigate an effect of the depth on the disposal depth. Through this study a sensitivity of a couple of varying depths for the disposal of SFs to the radiological safety is discussed in view of the radiological safety.

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Keywords: Spent fuel disposal, Deep geological repository, Disposal depth, Radiological safety

Considerations for the Bentonite Buffer Thickness in a Deep Geological Disposal of Spent Nuclear Fuels

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When establishing the concept and design of a buffer material in an engineered barrier of deep geological disposal system, the thickness of the buffer material is an important consideration. The buffer material serves (1) to prevent the intrusion of corrosive oxidants such as oxygen from the outside and to suppress the spread of nuclides leaking from the disposal container to the outside, (2) to effectively dissipate the decay heat generated in the disposal container to the outside, and (3) a function that can protect the disposal container from physical shocks such as external tectonic movements is required. If the buffer material is too thick, it not only makes it difficult to dissipate the decay heat, but also increases installation costs. And if the thickness of the buffer material is thin, the function of limiting the transfer of nuclides is reduced, and the physical buffering effect is also insufficient.

- Buffer Thickness due to rock shear displacement

In deep geological environments, buffer materials must be able to provide a physical buffer against tectonic movements. It is an important evaluation factor in countries with many earthquakes, such as Japan, and must also be considered in Korea, which is not completely safe from earthquakes. These criteria predict the maximum shear movement that can occur at a sufficiently deep geological disposal depth. In this paper, the maximum shear distance was set at 100 mm. The clearance between the disposal container and the ground almost disappears, so in order to have a sufficient safety factor, it seems that the thickness of the buffer material must be at least 200 mm or more.

- Buffer thickness according to nuclide release rate

When the disposal container is damaged and loses its function, the main mechanism through which the radionuclide leached by groundwater flows out to the bedrock around the disposal site through the buffer material is diffusion. KAERI evaluated the leakage rate through the buffer of 12 radionuclides known to be important nuclides in assessing the safety of high-level waste disposal. From these evaluations, the appropriate thickness can be said to be between 250 mm and 500 mm.

- Buffer thickness according to thermal conductivity

Typically, the thermal conductivity of the buffer material is lower than that of the host rock. Therefore, when the buffer material surrounds the disposal container in the deposition hole, the surface temperature of the disposal container increases accordingly. The temperature limit of the container is to be below 100°C in most countries. If the thickness of the buffer material with low thermal conductivity, which is thermally disadvantageous, increases, the temperature of the disposal container increases. In the preliminary calculation, when the thickness of the buffer material became 500 mm, the surface temperature of the disposal container began to exceed approximately 100°C. So, an upper limit on the thickness of the buffer material can be 500 mm.

From the results of the preliminary analyses on the thickness of the buffer material, it was preliminarily confirmed that appropriate performance could be obtained in a range of 20–50 cm in the thickness of the buffer material. Therefore, with these results, the thickness of buffer material should be optimized with the real disposal environmental data.

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Keywords: Spent nuclear fuels, Buffer thickness, Safety, Technical maturity, Engineered barrier, Geological disposal

Investigating Shear Behavior of Rock Joints in Granitic Rock Through Direct Shear Testing

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In the context of deep geological repositories for high-level radioactive waste, where waste materials are encased in a bentonite buffer within natural rock formations, the presence of rock joints poses a significant concern for the long-term safety of these repositories, spanning over hundreds of thousands of years. Understanding the behavior of rock joints on-site is crucial for ensuring the integrity of the repository throughout its planned duration. This research focuses on investigating the shear behaviors of rock joints within granitic rock through a series of direct shear tests. We utilized Hwang-Deung granite as the testing material. The effect of joint roughness on shear behavior of rock joints was investigated by preparing two types of rock specimens with flat and rough joints. Specimens with flat joints were constructed by overlapping intact rock blocks, while specimens with rough joints were generated through wedge splitting tests using larger intact rock blocks. Wedge splitting tests created rough joints in a tensile manner, resulting in joint roughness coefficients exceeding approximately 15. Subsequently, direct shear tests were conducted under four different vertical stresses to determine the shear strength parameters of the granitic specimens. Shear displacements were applied at a constant rate during the tests, and the joint profiles were visually examined using a 3D laser profiler before and after each test.

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Keywords: Rock joint, Direct shear test, Granitic rock, Joint roughness, Shear strength, Wedge splitting test

Damage-Coupled Hydro-Mechanical Modelling of Excavation Damaged Zone Development

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Drilling and blasting during the excavation of a deep geological repository results in the development of an excavation damaged zone (EDZ) in the host rock formation. The discontinuities within the EDZ weaken the mechanical properties of the host rock and act as preferential pathways for groundwater inflow. The estimation of the EDZ extent prior to construction considering disposal environment conditions is key to ensure safe disposal and for issuing countermeasures during the construction of the deep geological repository. As part of Task B-2 of the OECD NEA Horonobe International Project (HIP), this study aims to estimate the extent of the EDZ formation at the Horonobe URL after excavation and expansion at 500 m depth. To simulate EDZ formation, a damage-coupled hydro-mechanical model was developed using the FEM software COMSOL Multiphysics. The Mazars damage model was implemented to quantify the extend of mechanical during excavation, stress relaxation and lining installment. The Mazars damage model input parameters were obtained via uniaxial compressive strength tests and Brazilian tensile strength tests on 24 claystone sampled sourced from boreholes W-1, W-2, and W-3 of the Horonobe URL. Numerical analysis was conducted for the 350 m Niche No. 3 considering $S_H = 0.9 S_V$, to verify the developed model. Additional analysis was conducted for the 350 m Niche No. 6, considering $S_H = 1.3 S_V$, to examine the effects of different horizontal stress conditions. Preliminary estimations for water inflow were also conducted using the developed model. Based on the verified model, further numerical analysis was conducted for the planned 500 m Niche No. 8 for different horizontal stress conditions ($S_H = 0.7 S_V, 0.9 S_V, 1.3 S_V$) and the results were compared with the different estimations made by the Japan Atomic Energy Agency (JAEA) and the Nuclear Waste Management Organization of Japan (NUMO). The results highlight the capability of damaged-coupled models to estimate EDZ development and groundwater inflow.

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Keywords: Deep geological repository, Excavation damaged zone, Numerical modelling, Damage, Horonobe international project

Design of Bentonite Erosion and Piping Test (BEPT) in KAERI Underground Research Tunnel (KURT)

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High-level radioactive waste (HLW) is usually recommended to be disposed in deep geological repository which is placed over 500–1,000 m in depth. The deep geological repository is composed of engineered barrier system (EBS) and natural barrier system (NBS). EBS consists of copper-steel canister, bentonite buffer, bentonite mixed backfill, and low pH concrete plug. The bentonite buffer plays important roles for delaying groundwater from NBS, preventing canister from tilting or damaging, releasing decay heat from the spent fuel in canister. Bentonite buffer is directly contact with near field rock mass and groundwater inflows to disposal hole from the rock fractures. Groundwater inflow from the nearfield rock can erode the bentonite buffer blocks and make a water channel on their surface. These phenomena are referred as erosion and piping. These erosion and piping make the bentonite buffer lose its weight and it reduces the integrity of bentonite buffer and whole disposal system.

In KAERI, there are broad laboratory tests such as artificial fracture tests, free-swelling tests, and laboratory-scale piping tests. Based on lesson-learned in limited laboratory tests, field scale test which is called as bentonite erosion and piping tests (BEPT) is planned to demonstrate realistic interaction between bentonite buffer and near field rock mass. Site investigations were conducted for a few candidate sites to determine adequate site for BEPT. Determined site has sufficient capability of ground water flow and relatively shallow depth to control the test equipment.

Inflow borehole, test hole, and three monitoring holes are determined and cored in certain depth. Injection system and sampling systems are designed and packer system for each boreholes are designed. Bentonite thickness and arrangement of thermo-hydro-mechanical (THM) sensors are examined. Detailed design of BEPT will be updated until ends of 2024 and it will be installed in 2025.

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Keywords: Disposal repository, Bentonite, Erosion, Piping, Field test

Analysis and Design of Voloxidizer for Spent Fuel in Head-End

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A high-capacity voloxidizer which can handle several tens of kgHM/batch was developed to supply U_3O_8 powders to an electrolytic reduction reactor. To enhance the hulls and powders recovery ratio of the voloxidizer and remote operability in limited space, the reactor design should be optimized and modularized.

To determine its hull recovery ratio, the improved voloxidizer was subject to preliminary hull and powder separation testing and recovery ratio measurements using 53 cut zry-4 hulls ($\varnothing 9.5 \times L30$ mm), which correspond to 1 kg of spent fuel. Copper balls ($\varnothing 1$ mm, 110 g) were used as a substitute for the powder. The voloxidizer was modeled by using SolidWorks. For the modular design of the device, we selected the main components, namely the heater, utility, motor, reactor, valve, and structure, and the remote assembling and disassembling possibility of the selected components were analyzed in terms of such factors as visibility, interference, approach, and weight to obtain the modular design most suited for the target modules. The new double-reactor voloxidizer provides a mechanism for processing spent nuclear fuel and is capable of pulverizing rod-cuts using air and heat; its modular structure renders it easy to maintain and repair remotely, and the operators can remain far from the highly-radioactive hot cell. The modeled voloxidizer has the following components: a voloxidizer capable of processing 50 kg of spent nuclear fuel, reactor module in which the spent nuclear fuel is placed and oxidized, heating module that heats the reactor module to a high temperature, utility module to control the inner conditions of the reactor module, support module to support the heating module and reactor module, drive module to move the spent nuclear fuel within the reactor module, valve module to control the discharge of the spent nuclear fuel in the reactor module, and collecting-container module to collect the discharged spent nuclear fuel. The respective modules can be assembled and disassembled.

We selected the screw-and-rotation method for the double rotary drum to increase the hull recovery ratio. Through this approach, the pellets in the rod-cuts could be easily oxidized at 500°C , and after oxidation, the hulls could be separated using the backlash mechanism of the screw. Thus, we derived a voloxidizing mechanism for decladding spent fuel rods with high recovery efficiency by utilizing a double-reactor mechanism, wherein an internal reactor and external reactor is used to obtain stable oxide powder while simultaneously separating pellets and hulls.

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Keywords: Design, Voloxidizer, Oxidation, Spent, Fuel, Rod-cuts

Assessment of Changes in Buffer Properties in an In-Situ Engineered Barrier Experiment of Korea

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The In-situ Demonstration of Engineering Barrier System (In-DEBS) conducted a field test in tunnel of Korea Atomic Energy Research Institute starting from 2016 to evaluate the intricate thermal-hydraulic-mechanical dynamics of an engineered barrier within a high-level radioactive waste repository. Employing compacted bentonite blocks as buffer material, the In-DEBS experiment aimed to replicate real repository conditions. In 2022, the In-DEBS buffer underwent sequential dismantling, with subsequent collection and analysis of buffer samples to assess physical at various locations. Results revealed variations in the physical properties of the In-DEBS buffer, attributed to factors such as technical gap, dismantling procedures, and temperature gradients induced by heaters. Additionally, this study presents experimental comparisons between field-collected samples and untreated laboratory specimens. While empirical correlations based on laboratory data showed high accuracy, some disparities were noted between field sample properties and predicted values, underscoring the significance of field experimentation.

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Keywords: Bentonite, Thermal conductivity, Hydraulic conductivity, Swelling pressure, High-level radioactive waste

Experimental Study on Mineralogical and Thermal-Hydraulic-Mechanical-Chemical Properties of Bentonite Purified Using Column Froth Flotation

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Bentonite, known for its expansive nature, is a major candidate for use as a buffer material in engineered barrier systems. Recently, there has been an attempt to improve the purity of bentonite by selectively removing impurities using the froth flotation method, aiming to develop enhanced bentonite buffer material. This study investigates the mineralogical and thermal-hydraulic-mechanical-chemical engineering properties of purified bentonite achieved through column froth flotation. The X-ray diffraction analysis revealed an increase in the proportions of montmorillonite and a reduction in non-swelling minerals through column froth flotation, which is supported by both increase in cation exchange capacities and BET specific surface area. In addition, the study showed that the purified bentonite had an enhanced swelling ability and a competitive thermal conductivity satisfying the KRS⁺ criteria under a dry density condition of 1.6 g/cm³. Furthermore, an increase in Cs adsorption capacity of purified bentonite was also confirmed. This study provides valuable insights into the technical feasibility of cost-effective bentonite purification as a part of enhanced bentonite materials.

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Keywords: Montmorillonite, Column froth flotation, Thermal conductivity, Swelling pressure, Cation adsorption

Relationship Between Bentonil-WRK Bentonite Powder With Different Water Contents and Hyperspectral Imaging Data

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Bentonite is preferred as a buffer material in disposal sites by most countries due to its low hydraulic conductivity and high swelling potential, and is utilized in the form of compacted blocks to enhance performance. It is known that the performance of compacted bentonite buffer materials is influenced by the initial water content. Traditionally, the water content of bentonite was determined by comparing the weight before and after drying to derive the amount of water. However, this method possesses representativeness for the sampling location, necessitating the development of new methods for estimating water content. As part of this research, we aimed to analyze the correlation between hyperspectral imaging data and bentonite powder with varying water contents.

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Keywords: Bentonite, Buffer material, Water content, Hyperspectral image, Deep geological repository

Temperature Variation of Near-Field Rock in Deep Geological Repositories for High-Level Radioactive Waste According to Thermal Conductivity of Buffer

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To reduce the area of deep geological repositories for high-level radioactive waste (HLW), the thermal conductivity of the buffer must be improved while satisfying hydro-mechanical-chemical performance targets. As the thermal conductivity of the buffer increases, decay heat from the canister is more effectively transferred to the surrounding rock, necessitating an examination of temperature changes in the near-field rock due to the enhancement of buffer thermal conductivity. Therefore, in this study, the temperature changes in the near-field rock due to the enhancement of buffer thermal conductivity were investigated through numerical analysis using the finite element method. As the thermal conductivity of the buffer increases, the temperature of the near-field rock decreases. However, the rate of temperature reduction in the near-field rock due to the increase in thermal conductivity of the buffer was smaller than that in the buffer due to the increase in thermal conductivity.

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Keywords: Near-field rock, Buffer, High-level radioactive waste repository, Deep geological repository, Thermal conductivity, Rock temperature

Site Characterization for Bentonite Erosion and Piping Test (BEPT) in KURT

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Various safety measures are implemented for the secure disposal of high-level radioactive waste. Among them, bentonite serves as a buffer material placed between the disposal canister and the surrounding rock mass. Bentonite functions to keep groundwater from ingress and to facilitate the transfer of decay heat from spent nuclear fuel to the surrounding rock mass. The presence of hydraulic conductive joints or the formation of EDZ (excavation damaged zone) around the deposition hole can lead to erosion and/or the piping phenomenon in bentonite due to groundwater flow. It can significantly affect the effectiveness of bentonite as a buffer material so that comprehensive research on the phenomenon is necessary.

KAERI (Korea Atomic Energy Research Institute) has carried out many laboratory scale experiments to study the erosion and piping phenomena in bentonite specimens. For instance, an artificial joint, which was an ideal case in the form of flat and parallel plates, was used in an experiment to experimentally determine the erosion characteristics under various hydraulic and geochemical conditions. Based on the fundamental data and knowledge from those laboratory scale experiments, a field demonstration test, termed BEPT (Bentonite Erosion and Piping Test), is in the planning stage. The BEPT is planned to take place at the KURT (KAERI Underground Research Tunnel) located within the institute. It is of great importance to select an appropriate test site so that it can successfully achieve the research objectives. Therefore, various site investigations were conducted, which included geophysical logging, joint characterizations, and hydraulic tests. Through the investigations, a proper test site with proper borehole depth was selected. This paper aims to compile the investigation results and to present characteristics of the site.

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Keywords: Site characterization, Bentonite, Erosion and piping, Demonstration test

A Preliminary Study on Using Two-Region and Two-Site Models for Describing Non-Equilibrium Transport Process in APro

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Korea Atomic Energy Research Institute (KAERI) has been developing a process-based total system performance assessment model called APro. Within APro, the far-field module has employed the standard advection dispersion equation (ADE) to simulate solute transport. However, there is increasing evidences that the standard ADE inadequately captures transport phenomena in porous media. In order to improve this evolving model, KAERI is actively involved in Horonobe International Project (HIP) Task A, which aims to advance a realistic 3-D solute transport model that can be applied to safety assessments in fractured rock.

Prior to modeling of in-situ experimental results, a series of simulation benchmark tests is proposed to establish confidence in the utilization of each team's computational codes for describing complex transport processes relevant to demonstrating long-term safety of geological disposal repositories. The simulation benchmark tests encompass two conceptual models aimed at addressing the constraints of the standard ADE: (1) two-region model (representing physical non-equilibrium) and (2) the two-site model (representing chemical non-equilibrium). The two-region model, which is also referred to as dual porosity model, assumes that the porous media is composed of a mobile region (e.g. fracture) where advection takes place, and an immobile region (e.g. rock matrix) with stagnant water. Meanwhile, the two-site model assumes that sorption sites in porous medium can be classified into two fractions, with sorption processes occurring instantaneously at one fraction (f), and time-dependently at the remaining fraction ($1-f$).

In this study, two benchmark problems are modeled with COMSOL Multiphysics to investigate the viability of utilizing two-site and two-region model as potential mechanisms for overcoming the limitations of standard ADE within APro. The benchmark problems are approached in an inverse type, which aims to obtain similar breakthrough curve (BTC) as the given BTC through the fitting of parameters such as dispersion coefficient D and sorption parameter K_d . The modeling results indicate that two-region and two-site models can be effectively simulated using COMSOL Multiphysics. Particularly, the result of two-region model can be a foundation for the enhancement of the APro's numerical module for describing transport process in heterogeneous porous media.

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Keywords: HIP, Transport, Adsorption, Two-region model, Two-site model, COMSOL multiphysics, APro

Capability of the Numerical Analysis Model for the Near-Field Domain in APro

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APro (Adaptive Process-based total system performance assessment for a geological disposal system) presents a comprehensive framework tailored for simulating radionuclide transport influenced by thermal, hydraulic, mechanical, and geochemical variations within the disposal system. APro is currently focused on benchmarking the KBS-3V conceptual design for the disposal system, which can be divided into three distinct domains: the near-field domain encompassing the disposal canister, borehole, tunnel, buffer, backfill, and some surrounding rock; the far-field domain consisting of the remaining rock; and the biosphere domain including the upper strata and surface.

Given the distinct physical phenomena influencing radionuclide transport within each domain, APro's numerical analysis model was constructed with separate component models for each domain within COMSOL. The component models were then integrated by synchronizing boundary conditions to facilitate the total system performance assessment. To streamline the integration process, a hierarchical structure of physical phenomena was established, guiding the development of component models. Specifically, the basic phenomenon, belonging to the lowest layer, was designated as a 'module', and the optional models for each module were developed accordingly. The near-field domain includes models for 12 modules related to the transport of radionuclides and chemicals, groundwater flow, heat transfer, and safety function degradation. The far-field and biosphere domains include models for 9 and 5 modules, respectively, related to radionuclide transport, groundwater flow, and long-term evolution.

This study undertakes a case study to verify the capability of the numerical analysis model across all module combinations within the near-field domain. Each case undergoes rigorous scrutiny to evaluate numerical stability and accuracy, with an effort to explore strategies for enhancing them where necessary.

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Keywords: APro, Process-based total system performance assessment, Numerical analysis model, Near-field domain

Defining Vegetation and Intake Data of ADioS for Evaluating Long-Term Evolution Scenarios Using APro

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Given the current indeterminacy of the disposal site, the ADioS (Artificial Geological Disposal System) concept was adopted to develop the APro (Adaptive Process-based Total System Performance Assessment Framework) and conduct TSPA (Total System Performance Assessment) using APro. The formulation of long-term evolution scenarios for TSPA of ADioS draws on the geological characteristics and historical context of the Korean peninsula, complemented by climate modeling efforts conducted in collaboration with the University of Bristol. These scenarios depict changes in the surface environment through six repeating snapshots within each glacial cycle, capturing key variables such as air temperature, precipitation, sea level, and Köppen climate classifications in a representative snapshot. Additionally, specific snapshots were developed for tectonic uplift and global warming.

However, due to the lack of standardized vegetation types for Köppen climate classifications, our study aimed to define vegetation and intake data for snapshots used in the biosphere and dose assessment modules. We evaluated four methods: the HyTAG model, FAO Ecological Zones (FAO EZs), adjustments using the Korean peninsula climate distribution map, and the Korean peninsula vegetation distribution map to precisely match vegetation and intake with climate. This assessment helped determine the suitability of each method, and identify suitable ADioS-compatible vegetation and intake data.

Using the optimal vegetation and intake data obtained from this study, input data for SWAT in the biosphere module and dose assessment module that reflect the long-term evolution of the disposal environment on the Korean peninsula will be determined, allowing for the evaluation of the long-term evolution scenario.

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Keywords: APro, ADioS, Long-term evolution scenario, Snapshot, Vegetation data, Intake data

Preliminary Study on the Effects of Temperature and Radiation on Monitoring Sensors for HLW Repository

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Handling highly radioactive waste requires disposal systems known as high-level radioactive waste disposal systems. These systems are crucial for ensuring safety and efficiency, with monitoring sensors playing a key role. In this study, we conducted a preliminary experiment to understand how temperature and radiation affect the lifespan of monitoring sensors. This study conducted experiments targeting the accelerometer sensor, which is the most widely used across various fields among monitoring sensors. The experiment involved two groups of sensors; one group was exposed to radiation, while the others were not. Both groups were exposed to the same harsh environmental conditions, specifically a temperature of 170°C. We observed that sensors exposed to radiation are more likely to malfunction, highlighting the significant impact of radiation on the sensor durability. These findings are noteworthy because they help us better understand when both heat and radiation stress are applied to sensors and could improve sensor design and operational strategies in high-level radioactive waste disposal systems.

Acknowledgements

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Keywords: Monitoring sensor, Temperature, Radiation, High-level radioactive waste disposal system, Sensor durability

Evaluation of the APro Reaction Module by Benchmark Simulation for Cation Exchange Reaction

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The adaptive process-based total system performance assessment framework (APro) was developed at the Korea Atomic Energy Research Institute (KAERI) as a tool for the long-term safety assessment of a deep geological repository. The APro_Reaction module, one of the computational modules of APro, can evaluate detailed geochemical reactions such as mineral dissolution/precipitation, cation exchange, surface complexation and radionuclide sorption by coupling COMSOL and PHREEQC. A sequential non-iterative approach was used to couple the two software, and non-water H, non-water O and charge balance were included in the coupling processes to improve accuracy and stability.

In this study, the simulation results of APro were compared with other reactive transport models to verify the APro_Reaction module. PFLOTRAN, developed at Sandia National Laboratories in the United States, was used as the validation code and the same thermodynamic database from Thermochemie 12a was used to set up the geochemical reaction identically. The benchmark problem was set based on the results of column-scale cation exchange experiments in a previous study. As a result of the validation, APro showed results in good agreement with PFLOTRAN and experimental results. Future research will focus on validating the APro_Reaction module by performing benchmark problems of more complex geochemical reactions, such as mineral dissolution/precipitation, surface complexation and radionuclide reactions.

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Keywords: APro, Validation, Geochemical reaction, PFLOTRAN, Cation exchange

Procedure for Scenario-based Total System Performance Assessment by an Adaptive Process-based Total System Performance Assessment Framework (APro)

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The Adaptive Process-based Total System Performance Assessment Framework (APro) is a finite element method model designed to assess the transport of radionuclides in deep geological disposal systems. It also evaluates the resulting radiological dose experienced by a representative group of inhabitants living near the disposal site.

The numerical modules within APro facilitate the exploration of multi-physics phenomena within the disposal system, in conjunction with the transport of radionuclides across a three-dimensional domain that evolves over the long term. The long-term evolution of the disposal system is characterized by scenarios. These scenarios consist of qualitative descriptions detailing the multi-physics phenomena affecting material properties within the disposal system. Previous studies by KAERI have established scenarios for Total System Performance Assessment (TSPA), considering long-term climate changes.

In this study, we developed a comprehensive procedure for TSPA using APro and summarized assessment model flowchart (AMF) for the TSPA scenarios. This procedure assesses the transport of radionuclides and radiological dose by considering predefined scenarios. Within APro's TSPA procedure, scoping calculations are employed to select critical variables and evaluate input parameters essential for radionuclide transport. The models used in these scoping calculations are constructed by selecting appropriate numerical modules available in APro.

Based on the outcomes of the scoping calculations, an integrated model for radionuclide transport within the disposal domain is formulated for the TSPA scenario. This integrated model is constructed using APro's modules. Then, the radiological dose for the representative group, as defined by the scenario, is computed using the results obtained from the integrated model.

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Keywords: Geological disposal system, Total system performance assessment, AMF, APro, Long-term evolution, Scenario

Designing a Benchmark Problem and Hydrological Modeling Based on Biosphere Scenarios

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The Korea Atomic Energy Research Institute (KAERI) has developed the beta version of an adaptive process-based total system performance assessment framework for a geological disposal system known as APro- β . This framework compartmentalizes the entire disposal system and modularizes computational models to evaluate the complex phenomena within each compartment. As a component of APro, the biosphere module has been created to simulate the transport of radionuclides in groundwater flow and their release into surface water bodies, taking into account the long-term evolution of the surface environment.

This module is planned for validation through code comparison with Hydrogeosphere (HGS), a state-of-the-art 3D finite element simulator used to model water and solute movement within watersheds. To facilitate this comparison, the study has defined a benchmark problem for a synthetic site called the Artificial Disposal System (ADioS), based on biosphere scenarios that depict seven different snapshots (SN0, SNE0, and SN1~5) of a 100,000-year glacier cycle. The objective is to assess hydrological conditions representing these eight independent snapshots.

The biosphere scenarios provide datasets on temperature, precipitation, sea level, and Köppen climate classification for each snapshot. These datasets are extensively utilized as inputs for the biosphere module, which undergoes two modeling steps to simulate hydrological conditions: water balance analysis and groundwater flow simulation. In the first step, the Soil and Water Assessment Tool (SWAT) simulates the normal water balance components for each snapshot, including evapotranspiration, groundwater recharge, and surface water table. In the second step, the previously calculated data are used as boundary conditions to simulate steady-state groundwater flow and discharge, which are key drivers of radionuclide transport.

The results demonstrate how the developed APro's biosphere module computes representative hydrological conditions for each snapshot. These results are intended to be compared with those simulated by HGS for the same benchmark problem to validate the module and ensure its quality.

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Keywords: APro, Biosphere, Water balance, Groundwater flow, Benchmark test

Improvement and Verification for Near-Field Non-Isothermal Two-Phase Flow Model in APro

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In recent years, due to the demand for the process-based performance assessment of a geological radioactive waste disposal, beta version of APro (Adaptive Process-based total system performance assessment framework for a geological disposal system) was proposed by Korea Atomic Energy Research Institute (KAERI). In APro beta version, previously developed non-isothermal two-phase flow model for near-field simulations did not address the behavior of vapors in detail because they considered repository conditions dealing with temperatures below 100°C. However, recent research efforts are aiming to increase the maximum temperature in the bentonite buffer to over 100°C, which could potentially affect the migration of gas-phase radionuclides due to vapor movement, and therefore, a model capable of considering vapor diffusion is needed for the performance assessment. In addition, gas pressure and gas saturation were used as primary variables in the existing model. The advantage of this scheme lies in the easy handling of capillary pressure from a modeling perspective, achieved by treating capillary pressure as a secondary variable. However, this scheme cannot model the discontinuity of saturation occurring between two different media, which makes it impossible to accurately simulate the hydraulic behavior of heterogeneous media in the early stage of disposal. Therefore, as an alternative, a scheme considering capillary pressure as the primary variable has been proposed. With this scheme, it is possible to model the continuity of capillary pressure between two media with different properties and simulate the discontinuity of saturation, thus allowing for a more accurate representation of hydraulic behavior in heterogeneous media.

In this study, the existing non-isothermal two-phase flow model was improved to use gas pressure and capillary pressure as the primary variables in the governing equations and to consider the diffusion of vapor in simulating gas movement. Furthermore, to verify the improved model, numerical simulations were performed under various thermal, hydraulic and mechanical conditions and the results were compared with the analytical solution. The validated non-isothermal two-phase flow model will be used as part of APro in integration with the radionuclide transport model to evaluate the migration of radionuclide considering the effect of coupled thermal-hydraulic-mechanical (THM) processes in the near field.

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Keywords: High-level radioactive waste repository, Process-based total system performance assessment, APro, Non-isothermal two-phase flow model, Verification

Generation and Stability of Colloids From Compacted Bentonite Under the Synthetic Groundwater Condition

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The spent nuclear fuel disposal strategy under consideration within domestic frameworks involves encapsulating spent nuclear fuel within metal canisters, which are subsequently interred in the deep geological repositories carved into crystalline bedrock. These canisters are ensconced within a matrix of bentonite, a material chosen for its buffer qualities, including favorable physical properties and an elevated capacity for nuclide sorption. Nonetheless, this strategy is not without its vulnerabilities; groundwater intrusion into the repository can provoke the erosion of the bentonite buffer, leading to the genesis of colloidal particles. Due to their size, these colloids have the potential to facilitate the expedited migration of radionuclides, thereby posing a challenge to the containment integrity of the disposal system. Given the extensive temporal horizon envisioned for the disposal of spent nuclear fuel, it becomes imperative to conduct a thorough assessment of both the volume of colloids that could originate from the buffer material and the stability of these colloids, as such parameters are integral to the comprehensive evaluation of disposal system safety. Therefore, this study developed an experimental apparatus designed to simulate the disposal environment, specifically focusing on the chemical erosion of compacted bentonite and the resultant generation of colloids. The experimental investigations employed two varieties of bentonite: Ca-type Bentonil WRK and Na-type Wyoming bentonite, the latter being widely utilized on an international scale. The bentonite samples were compacted to a density of 1.6 g/cm³ within an acrylic system featuring artificial fractures of 1 mm, simulating the potential for groundwater ingress. The outcomes of these experiments indicated a more pronounced colloid production from Wyoming bentonite when compared to Bentonil WRK, a discrepancy likely attributable to variations in mineralogical composition and granularity between the two types of bentonite. Observed colloidal particles exhibited an average size range of approximately 300 to 400 nm, with experiments conducted under conditions of high ionic strength groundwater revealing a tendency towards reduced colloid generation and increased aggregation, attributed to the reduced stability. This research elucidates the variable nature of colloid generation contingent upon the specific bentonite type, alongside the influence of the infiltrating water characteristics such as ionic strength on colloid stability. The findings of this investigation are anticipated to serve as foundational data for future analyses aimed at elucidating the dynamics of colloid and nuclide migration acceleration originating from buffer materials.

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Keywords: Colloid, Bentonite, Chemical erosion, Colloid-facilitated migration, Stability

Radionuclide Transport Modeling Considering Adsorption in Excavation-Damaged Zone

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High-Level radioactive Waste (HLW) is typically managed via deep geological disposal, a process that entails isolation of HLW within a stable geologic formation. In order to ensure long-term safety of the disposal, a multi-barrier system is employed to prevent the leakage of radionuclides, such as U(VI). One of the factors that can compromise the long-term safety of the barrier is the Excavation-Damaged Zone (EDZ), which is a fractured zone generated around excavation area.

Within the EDZ, the fracture network can serve as a pathway for groundwater infiltration to the barrier. According to several researches, when the bentonite contacts groundwater, there is a possibility of bentonite extrusion into the fracture network. In this case, U(VI) can adsorbed to the surface of the extruded bentonite. Therefore, considering extrusion length is crucial for evaluating how extruded bentonite can affect the U(VI) leakage. To simulate this, coupled hydro-geochemical modeling was conducted progressively changing extrusion length of the bentonite. The model consists of both COMSOL Multiphysics, used to simulate U(VI) transport, and PHREEQC, employed to calculate adsorption of the U(VI) on the surface of the extruded bentonite.

According to the results, U(VI) was transported preferentially through highly connected fractures, facilitated by the rapid flow of groundwater. Along the preferential pathway, the behavior and retardation of U(VI) exhibited significant changes depending on the extrusion length of the bentonite. As the length of the extrusion length increases, there is a corresponding increase in the amount of U(VI) adsorption. These results indicate that property changes of EDZ has a significant effect on the behavior of U(VI). In further studies, investigations will be conducted to examine the changes in preferential pathways by varying distribution of fractures and connectivity.

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Keywords: Excavation-damaged zone, Uranium, Bentonite extrusion, Reactive transport, COMSOL multiphysics, PHREEQC

Thermodynamic Modeling of Calcium-Sodium Exchange Reactions in Bentonil-WRK Montmorillonite

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Deep geological disposal (DGD) has been developed and is considered the safest engineering option for isolating high-level radioactive waste (HLW), such as discharged spent nuclear fuel (SNF), from biosphere until its radiological risks reduce to the level of natural uranium. The DGD system consists of multiple barriers, where bentonite clay-based buffer and tunnel backfill are key engineered barriers designed to prevent the ingress of groundwater and chemicals that compromise the safety of waste canister and to retard the migration of released radionuclides by providing sufficient chemisorption sites. Montmorillonite, a swellable 2:1 phyllosilicate, is the primary mineral component of bentonite, and it predominates the hydrogeochemical performances of the clay. Bentonite and montmorillonite are generally categorized into Na- or Ca-type according to their interlayer cation composition, and Ca-type clays are being considered as candidate materials for the engineered barriers of deep geological repository (DGR) for HLW in numerous countries; compressed Ca-bentonites have comparable swelling capacities to Na-bentonites at high dry density while they generally possess economic advantage and may display superior engineering performances in terms of thermal conduction and anion sorption. In 2021, Korea Atomic Energy Research Institute (KAERI) introduced a new reference research Ca-bentonite, Bentonil-WRK, and its thermal-hydrological-mechanical-chemical (THMC) properties are under investigation in-depth. Especially, sorption of highly soluble fission products onto Bentonil-WRK montmorillonite has been examined under a wide range of geochemical conditions, and associated thermodynamic data which can improve predictability in demonstrating the retention capacity of clay adsorbent compared to that of utilizing conventional K_d (sorption-desorption distribution coefficient) values were evaluated. Within this framework, we aimed to assess and thermodynamically model calcium-sodium exchange reactions in the clay adsorbent. Such reactions are key geochemical processes that occur upon seawater intrusion into the engineered barrier system, and thus obtaining relevant thermodynamic data is essential for accurately evaluating the long-term evolution of geochemical environment surrounding the DGD system. The present work summarizes the utilization of thermodynamic sorption modeling (TSM) and characterization of purified Na- and Ca-montmorillonites from Bentonil-WRK bentonite. Additionally, calcium-sodium exchange behaviors of the montmorillonite at a temperature of 25°C were examined through batch experiments under various ionic strength conditions ($I = 0.01\text{--}0.1\text{ m NaCl or CaCl}_2$), and the resulting thermodynamic data are discussed accordingly.

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Keywords: Engineered barrier, Montmorillonite, Calcium, Sodium, Ion-exchange, Thermodynamic sorption modeling

Iodide Retention Mechanisms on Oxidized Cu Canister Under Expected Repository Conditions: Insights Into High-Level Radioactive Waste Disposal

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Radioactive iodine with a long half-life, produced as a fission product from uranium in spent nuclear fuel, poses significant environmental risks. Specifically, ¹²⁹I is particularly hazardous due to its radiotoxicity to humans and high mobility in the subsurface environment. This study elucidated the processes and mechanisms of iodide retention, focusing on CuI_(s) formation facilitated by the oxidation of a copper canister under saline conditions typical of deep geological repositories. We also provide a comprehensive understanding the iodide retention processes by copper canister, based on the results from several analytical techniques such as XRD, AFM, TEM/SEM, XPS, and geochemical modeling. In the event of canister failure, radioactive iodide ions can be stably retained, and their migration can be effectively hindered by CuI_(s) formation on the surface of oxidized copper canister. These findings suggest that the copper canister functions as a physicochemical barrier, impeding the radioactive iodide migration in such repositories.

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Keywords: High-level radioactive waste, Iodide, CuI_(s), Retention, Saline, Copper canister

Geochemical Evaluation of Mafic Dykes in KURT Site: Implications for Long-Term Evolution of a Deep Geological Repository

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The global consensus on the disposal of high-level radioactive waste advocates for a multi-barrier system, integrating engineering and natural barriers for assured containment. The Korea Atomic Energy Research Institute (KAERI) is at the forefront, validating the safety of such a system within the KAERI Underground Research Tunnel (KURT). KURT is in a bedrock composed of Jurassic granite, and because of this, lithologic studies at the KURT site have been conducted mainly on granitic rocks, and the importance of mafic dyke has been overlooked. This study introduces fresh core samples from KURT site, combining earlier research findings to propose a methodology that examines the long-term stability and evolution of the natural barrier.

To consider how rock geochemistry can be applied to the study of long-term evolution of disposal site, we analyzed the geochemical composition of bedrock and dyke, utilizing cores from four newly drilled boreholes (GE-1 through GE-4) at depths nearing 200 meters. Our comprehensive geochemical investigation encompasses major and trace elements, alongside Sr-Nd-Pb isotopic analysis, to decipher the rocks' petrogenesis. Additionally, we aim to employ zircon U-Pb and whole-rock Ar-Ar geochronology to determine the rocks' ages. This study also accumulates geochemical data from prior explorations of the KURT site and the Yuseong area.

Preliminary findings suggest that the mafic dyke's presence is indicative of Cretaceous volcanic disruptions in the Yuseong area, with geochemical signatures hinting at a subduction-related tectonic environment. A thorough geochemical examination of the host rock and dyke promises deeper insights into the natural barrier's long-term evolution, potentially yielding qualitative and quantitative benchmarks for safety assessment.

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Keywords: KURT, Natural barrier, Granite, Dyke, Geochemistry, Geological evolution

Benchmarking of K-BANDI for Groundwater Release Into Agricultural Environments in a Constant Biosphere System

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RADCORE is developing a biosphere assessment integration platform program for Korean generic deep geological repository safety case based on GoldSim, named K-BANDI (Korean-Biosphere system Assessment code for spent Nuclear fuel Deep geological Isolation). As a part of verification of K-BANDI ver. 1.0, a benchmark calculation was undertaken with an approach in the NCRP-129, which can be applied to screening sites where the surface soil is known to be contaminated with radionuclide. After obtaining the respective simulation to equilibrium results for the scenario of farming irrigated with surface water using different biosphere assessment conceptual and mathematical model whilst identical input parameters of ingestion rate, inhalation rate and external exposure times, they were compared each other. Exposure pathways considered for river water medium were ingestion of drinking water, freshwater fish, terrestrial crops in the irrigated soil, meat and milk, irrigated soil, inhalation of resuspended soil, and external radiation by irrigated soil. In this benchmarking study, biosphere system change was not considered consistent with the assessment context specified for the development of K-BANDI ver. 1.0. While the K-BANDI dynamic compartment models based on the mathematical representations in IAEA BIOMASS were implemented within the GoldSim, the NCRP-129 static or equilibrium models were implemented in a MS Excel spreadsheet. The assessment endpoint used for the comparison is the Biosphere Dose Conversion Factor (BDCF), which is defined as the maximum value of the total dose to the potentially exposed group, in Sv/yr, resulting from a continuous unit release of 1 Bq/yr during the whole simulation time to the river compartment. In spite of considerable differences in the element-dependent data such as distribution coefficients in soil, soil-to-plant concentration ratios, and transfer factors to plant and animal products etc., the results are within a maximum difference of 4 orders of magnitude. Except for some of the actinide nuclides such as Np-237, Cm-244, Pu-240, Pb-210, and U-235, the differences are within 2 orders of magnitude. In particular, for actinide nuclides which generally have low rates of root uptake, contamination due to direct deposition of irrigation water or resuspended particles on plant surfaces would be important. Differences in input parameters associated with resuspension processes from the soil (e.g., in the K-BANDI, the transfer factor for actinide nuclides was around 0.045 to 0.3, but in the NCRP-129 model, it was applied as 0.001.) are thought to be a main cause. Overall, the K-BANDI calculation results for most nuclides except some actinides show similar BDCFs with those from the NCRP-129 approach.

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Keywords: Biosphere assessment, Mathematical model, Surface water farming, Dynamic model, Equilibrium model

Development of Sorption and Diffusion Database Management System for the Safety Assessment of Radioactive Waste Disposal

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Deep geological disposal is mainly considered worldwide for the final disposal of spent nuclear fuel. The safety assessment of deep geological disposal system requires various input parameters. Among the input parameters, geochemical behavior of radionuclides in deep geological environment is an important factor in safety assessment, which include migration and retardation processes such as dissolution and formation of aqueous species, precipitation/co-precipitation, advection/dispersion, sorption and, diffusion etc. Sorption and diffusion are main processes in retardation of radionuclides in geosphere. Sorption and diffusion are mainly affected by geological media, solution chemistry, and radionuclide chemistry in a complicated way. Hence, these retardation processes are very dependent in the experimental conditions. So, distribution coefficients and diffusion coefficients are managed in the database management system or reports in many countries. KAERI-SDB^a ver. 1.0 and KAERI-DDB^a ver. 1.0 have been developed for utilizing it as supplementary data for safety assessment. Input data for DB management system consists of four main components; experimental data, solution chemistry, solid information, and reference sheet. They will be released on the web in 2026, allowing stakeholders involved in radioactive waste disposal and general public to conveniently seat and extract data.

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Keywords: Sorption, Diffusion, Database, Safety assessment, Deep geological disposal

Development of Improvement Plan for Safety Assessment Complex Scenario Using Goldsim

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In the concept of deep geological disposal of domestic high-level radioactive waste, the disposal site must isolate and delay the movement of radionuclides for hundreds of thousands of years after closure. At this time, a disposal safety assessment must be conducted to quantitatively evaluate the radiological risks to residents and the natural environment living near the disposal site due to site characteristics and multiple barriers. Disposal safety assessment is largely because it takes a considerable amount of time for radionuclides still contained in the disposed waste to escape from the disposal site and be exposed to the human ecosystem again due to various variables such as earthquakes, climate, and human invasion after a long period of time. Considering the characteristics of nuclides being attenuated or captured or adsorbed to other substances along the way, the purpose is to provide evidence that they will not be affected by radiation and will be safe even if exposed after a long period of time.

Disposal safety assessment is divided into operational safety assessment and post-closure safety assessment. Operational safety assessment evaluates the radiological risks to workers and nearby residents in the process of transporting, handling, and disposing of high-level waste while the deep disposal site is in operation. The post-closure safety assessment is a long-term evaluation period spanning hundreds of thousands of years after the deep disposal site is closed. The techniques used and the phenomena considered are different from the safety assessment of other nuclear facilities. The post-closure safety assessment is conducted at the deep disposal site. This is a major process that quantitatively confirms intrinsic stability.

In order to perform such a disposal safety assessment, various procedures such as scenario design and radiological impact calculation are required, and the selection of a conceptual model and mathematical model for each element is required by considering site characteristics and natural environmental factors. There are various programs for performing or interpreting actual disposal safety evaluation, representative examples include AMBER, MASCOT-K, GENII, GWSCREEN, and Goldsim.

Among them, Goldsim is widely used in safety evaluation of radioactive waste disposal not only in various countries but also in Korea. In the case of Goldsim, it is an optimal program development tool for simulating mass and volume cutting within complex disposal systems, such as nuclide movement in various media and artificial and natural barriers or groundwater flow. A detailed evaluation of the ecosystem is possible by considering leak scenarios from a more diverse and realistic perspective than previous models through screening of various Features, Events, and Processes (FEPs) involved in the disposal system.

In this study, we investigated the definition of disposal safety assessment and various programs accordingly. In addition, the existing safety evaluation scenario was reconstructed using goldsim, and improvement measures were analyzed. If the construction of a high-level radioactive waste repository is started in the future, it is expected that the disposal system, including the geological environment and groundwater flow and conditions, will need to be accurately identified for further safety assessment, and sensitivity analysis of various scenarios and inputs used in the program will need to be performed.

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This paper is a result of The Human Resources Development Project for HLW Management hosted by KORAD.

Keywords: Goldsim modeling, Safety assessment, HLW, Radioactive waste

Using Goldsim to Create Concrete Deterioration Scenarios in Radioactive Waste Disposal

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Concrete plays a very important role as a repository material in the domestic high-level radioactive waste core disposal environment. Concrete is one of the most important engineering barrier materials in the multi-barrier concept of radioactive waste disposal. In the radioactive waste disposal environment, it loses its role as a barrier due to various deterioration phenomena such as waste dissolution, groundwater flow, subsidence, and earthquakes, resulting in leakage of disposed waste into the natural ecosystem. A representative factor of concrete deterioration is sulfate-induced deterioration, which is the most common deterioration phenomenon when concrete structures are in contact with groundwater or water.

A radioactive waste disposal facility must be shown to be radiologically safe after closure. To show this, it is necessary to analyze the possible phenomena that may occur in the disposal facility and the surrounding environment after closure. However, it is not practical to predict these phenomena, so the program can be used to check the predictable phenomena and scenarios.

There are many different post-closure safety assessment programs. One such program is Goldsim, which was used for the licensing of the Yucca Mountain repository in the United States and is the best program tool for simulating mass and volume transfer in complex disposal systems, such as nuclide migration across artificial and natural barriers and groundwater flow. It has also been used to evaluate concrete deterioration, which has been studied extensively.

Prior to deriving scenarios using Goldsim, a concrete deterioration evaluation using KURT groundwater and sulfate was conducted to confirm the concrete deterioration phenomenon as a preliminary research experiment. For the experiment, a water tank was built and concrete specimens were immersed in it for a total of 365 days to accelerate concrete deterioration. After removing the specimens at 30-day intervals, characteristic analyses such as SEM and EDS were performed to confirm the concrete deterioration phenomenon.

In this study, basic data were obtained through preliminary experiments to derive concrete deterioration scenarios, and a basic evaluation model using Goldsim was presented. As a basic evaluation model, we selected a scenario in which waste disposed of in the repository leaks into nature through the disposal container, silo concrete barrier, and natural barrier and leaks into the ecosystem. In the future, data such as the amount of nuclide material, half-life, and artificial barrier characteristics necessary for safety evaluation will be secured and evaluated.

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This paper is a result of The Human Resources Development Project for HLW Management hosted by KORAD.

Keywords: Goldsim modeling, Concrete deterioration, HLW, Radioactive waste, Post-closure safety assessment, FEP

Enhancing Disposal Efficiency: Criticality Analysis of Rod Consolidation for Spent Nuclear Fuel in Deep Geological Repository

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Interest is growing in efficient pre-disposal treatment methods for high-level radioactive waste (HLW) such as rod consolidation and oxidation processing. Particularly in Korea, where securing disposal sites is challenging hence the anticipated costliness, maximizing disposal efficiency is paramount. Among various methods being researched to enhance disposal efficiency, rod consolidation, which involves physically disassembling nuclear fuel assemblies to densely pack them, is proposed as one solution. Rod consolidation becomes more attractive as the disposal facility with elevated temperature criteria has been discussed recently. Since multiple assemblies are densely stored, concerns regarding criticality need to be addressed. This study examines the feasibility of rod consolidation from a criticality safety perspective considering scenarios under both normal conditions and potential accident events, including copper canister failure. Criticality analyses were conducted utilizing MCNP6.2 for Korean pressurized water reactor (PWR) spent nuclear fuel assuming various uranium enrichments (3–6w/t%), burn-ups (30–70 GWD/MTU), and packing ratio (2:1, 3:2) in a Deep Geological Repository (DGR) setting. The various inventories of spent nuclear fuel were generated using SCALE 6.3.1 employing ORIGEN binary data from KAERI for Korean PWR. Adopting a 2:1 rod consolidation approach led to a criticality increase of about 1.5 times compared to an unconsolidated canister setup, with criticality levels maintained below 0.4. In the event of a copper failure, water entry replaces helium gas within the canister. The denser configuration of spent fuel rods substantially limits the escalation of criticality, due to the decreased moderation effect of water. In all modeled scenarios, including those with low burn-up and high enrichment, the results satisfied the regulation that criticality should not exceed 0.95. This indicates that rod consolidation technology maintains criticality safety standards and may even enhance the criticality safety profile in case of water entering the canister.

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Keywords: Rod consolidation, Critical analysis, Deep geological repository, Quantitative analysis, MCNP6.2

Processes of Long-Term Changes in the Surface Environment Due to Natural Climate Change, Uplift-Subsidence and Erosion-Deposition

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The performance and safety evaluation period of the deep disposal system covers a very long period of time. During such a long period, the surface environment can change due to natural changes in the climate and internal activities of the Earth such as uplift and subsidence. In this study, the evolution process of the surface environment due to the natural phenomena is presented in a flowchart.

In the process of changing the surface environment by natural changes in the climate and internal activities of the Earth, the components of the surface environment directly affected by both factors were determined to be changes in sea level and resulting changes in topography. Based on the two components, the flowchart showing the surface environment change process due to the two factors was comprehensively organized.

Using each flowchart representing process of changes in the surface environment by natural changes in the climate and internal activities of the Earth, the process of changes in the surface environment under the condition where both factors are involved together was presented in the form of a flowchart. The two components play role as joints of two each flowchart.

The flowcharts qualitatively presented by integrating influence and change processes due to natural changes in the climate and internal activities of the Earth, which are factors changing the surface environment, will be used to evaluate the long-term stability of domestic sites for deep geological disposal.

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Keywords: Deep geological disposal, Climate change, Uplift, Subsidence, Surface environment, Long-term change

Characteristics of Alteration in Jurassic Granitoids Using Boreholes Around KURT Site

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Disposal facilities for high-level radioactive waste should be located to provide isolation from the accessible biosphere for hundreds of thousands to a million years after closure. In the concept of geological disposal, granite is internationally considered a favorable rock-type for maintaining such disposal safety performance. While granite has advantages of high geological stability and low permeability, the disadvantage of granite is that groundwater flowing through fractures generated within solid rocks can transport radioactive materials. Therefore, it is essential to understand the distribution of geological structures and the mineralogical and geochemical properties around the structures that can affect nuclide behavior for assessing long-term safety of the repository. Moreover, their long-term evolution and possible changes over the period of interest should be considered for long-term safety of the natural barrier performance. In this study, we investigated possible alteration structures in Jurassic granitoid of South Korea using the boreholes around KURT area as a test site. Highly altered rocks were distinguished from host granitoids, and their chemical and structural differences were compared. The distribution and properties of the minerals filling and coating fracture surfaces that can affect the nuclide behavior have been investigated. This is a preliminary study for the construction of alteration and lithological model around KURT area. The results will be applied for future studies on the long-term evolution of hydraulic-geochemical performance of natural barrier and long-term evolution of geochemical performance around the KURT site. In addition, the results will contribute to developing a methodology to construct a long-term geological evolution history of the natural barrier.

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Keywords: Geological disposal, Natural barrier system, Long-term evolution, Alteration, Deep crystalline rock

Evaluation of Diffusion Characteristics of Radionuclides in Buffer Matrix Under Oxidation Conditions

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A multi-barrier system consisting of canisters, buffers, backfill, and natural rock is considered the most effective way to isolate high-level radioactive waste and has been studied for several decades. Among these components, the buffer, which is a crucial material of the engineered barrier, plays a main role in retarding the release of radionuclides from the canister. The retardation mechanism in the deep geological environment is divided into sorption and diffusion. This study analyzes the diffusion properties of radionuclides in bentonite, a common buffer matrix, and evaluates the diffusion coefficients under oxidation conditions for the purpose of safety assessment for the deep geological disposal system. The radionuclides included in this study are alkali and alkaline earth metals (Cs, Sr, Ba), lanthanides (Sm, Eu), actinides (Ac, Am, Cm, Np, Pa, Pu, Th, U), transition elements (Nb, Ni, Pd, Tc, Zr), and others (C, Cl, I, Rn, Se, Sn). The diffusion of radionuclides is largely affected by the geochemical properties of the buffer and pore water, as well as the reaction characteristics of the radionuclides. For the evaluation of diffusion coefficients, data collected from Sweden (SKB) reports were analyzed, and recommended diffusion coefficients were suggested. The results of this study are expected to be utilized in determining the migration and retardation factors of radionuclides in buffer matrix for the safety assessment of a deep geological disposal system.

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Keywords: Bentonite, Groundwater, Radionuclide, Distribution coefficient, Deep geological repository

Investigating the Distribution of Radionuclides in Sedimentary and Granitic Aquifers Based on Age Tracers, Stable Isotopes, and Microbial Community Distribution

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To understand the factors determining the distribution of radionuclides, we analyzed the groundwater samples for age tracers, isotopes, and microbial community distribution. The use of age tracers revealed a tendency for increased radionuclide mobility in areas where infiltration of young groundwater was identified, indicating the impact of groundwater residence time on radionuclide migration. Analyzing the stable isotopes identified the retardation characteristics of radionuclides under sulfate reducing conditions, highlighted the importance of redox conditions over the radionuclide distribution. Microbial community distribution was useful to indicate the redox transitions in radionuclide migrating pathways. In granitic aquifer, the infiltration of young and oxidizing water was found dominantly around fractured zones while, in sedimentary aquifer, the mixing was noticeable around groundwater discharge zones near stream. This research demonstrates the usefulness of age tracers, isotopes, and microbial distribution in explaining radionuclide transport and retardation phenomena in various natural barrier system.

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Keywords: Radionuclide, Sedimentary, Granite, Age-dating, Stable isotope, Microbial community

Demonstration Test of Emplacement and Retrieval Devices for Spent Nuclear Fuel in DBD

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This study focuses on empirically testing emplacement and retrieval devices for spent nuclear fuel within the Deep Borehole Disposal (DBD). The verification testing process revealed deficiencies in the initial stage's emplacement/retrieval device, particularly in its control and safety mechanisms, guiding and connecting components, and the insertion/fastening section. Although a design improvement plan was devised based on empirical testing, it was evident that fundamental issues, such as the increased complexity arising from integrating both emplacement and retrieval functions into a single device, remained unresolved.

Consequently, the objective of integrating both functions into one device was reevaluated, leading to the development of separate emplacement and retrieval devices, each tailored to its specific function. A scaled-down model of a test device was conceptualized to demonstrate the emplacement/retrieval of simulated disposal canisters in a borehole with a 10 cm inner diameter.

Furthermore, structural integrity evaluation using a computer program, confirmed the structural safety of all components, considering the lifting conditions of the DBD emplacement /retrieval device. The designed emplacement/retrieval device comprises a connection device to link disposal canisters in a line or attach them to the lifting device connector, employing a simple mechanically operated connector rather than an electrically operated one. The connection device disconnects by releasing the hooks holding the disposal canister upon reaching the borehole's bottom.

To develop an effective emplacement/retrieval device for spent nuclear fuel within the DBD system, a demonstration test was conducted in a 10 cm diameter borehole. Using a scaled-down model simulating the actual device, weaknesses and problems identified in preliminary tests were addressed in the production of the demonstration test device. The successful emplacement and retrieval verification test of the disposal canister using the design-improved demonstration test device signifies a significant step forward in advancing DBD technology.

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Keywords: DBD, Spent nuclear fuel, Emplacement, Retrieval

Engineering-Scale Tracer Tests in a Natural Fractured Granite Rock

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The long-term stability of deep geological disposal systems for radioactive waste can be secured based on a comprehensive understanding of how the leaked radioactive nuclides are transported through and impeded by multiple barriers, including bentonite buffer and underground bedrock. Considering the complex nature of physico-chemical interactions in the heterogeneous geological media, our study examined the migration mechanisms of nuclides by conducting preliminary tracer tests in a natural granite rock block with a well-defined fracture. The aperture distribution and the mineral compositions of the rock in the vicinity of the fracture were analyzed and followed by the migration experiments using non-radioactive nuclides (Cs, Sr, Ni, Cl, I) with deuterium water (D₂O). Our preliminary results showed the acceleration and delay characteristics of the nuclide transport depending on their chemical properties. The migration of non-sorbing Cl ions was accelerated due to the anion exclusion effect on the negatively charged fracture surfaces. However, the migration of the strong-sorbing cationic ions (i.e., Cs, Sr, Ni) exhibited a significant delay effect.

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Keywords: Tracer test, Fractured rock, Nuclide migration, Acceleration and delay characteristics

Assessing the Influence of Chemical Composition of Solution and Temperature on the Zeolitization of Montmorillonite Using Natural Analogue Samples

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Zeolites typically form through the sedimentation of volcanic glass and the alteration of montmorillonite under geoenvironmental conditions characterized by high pH and Si/Al ratios. The zeolitization process entails a structural transformation of montmorillonite from a 2:1 layer structure to a framework of chain or stacked chains. Consequently, montmorillonite, a primary constituent mineral in bentonite, undergoes transformation into zeolite, which can influence the primary properties of buffer materials such as swelling capacity and low hydraulic conductivity. In deep geological disposal systems, there is a possibility of creating higher pH environmental conditions due to the contact between concrete reinforcement materials and groundwater. The interaction between high pH groundwater and buffer materials may lead to the zeolite formation, thereby affecting safety considerations. At the natural analogue study site, outcrop samples exhibit mineral compositions similar to those of typical bentonite and no zeolites were identified in outcrop samples. In contrast, drilled core samples from most of depths reveal predominantly zeolites, and montmorillonite was only weakly observed at a few drilled core samples. It is considered to have been influenced by the chemical composition of the precursors, geoenvironmental conditions, and various factors affecting mineral formation during geological processes. To investigate the factors influencing zeolite formation under geoenvironmental conditions, experiments were conducted utilizing bentonite outcrop samples to identify the chemical composition, concentration, and temperature of the reaction solution simulating a specific geoenvironmental condition.

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Keywords: Zeolite, Bentonite, Geoenvironmental condition, High pH, Framework of chain

A Preliminary Study on the Applicability of Reactive Transport Model to Uranium Transport in Single-Well Push-Pull Test

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To assess the safety of high-level radioactive waste disposal, it is essential to explore the uranium mobilization in association with both geological and geochemical environments. The transport of uranium in groundwater is sensitive to adsorption, precipitation, and dissolution processes depending on geochemical environments. However, it is challenging to couple groundwater flow and geochemical models due to the complexity of non-linear chemistry calculations. To integrate complicated chemical reactions in groundwater flow, we couple COMSOL Multiphysics and PhreeqcRM models. In this study, we perform preliminary reactive transport modeling based on field investigation in a test site to investigate uranium transport depending on the redox condition.

The study site is composed of uranium-bearing black slate in the Okcheon metamorphic belt. Despite the presence of uranium-bearing formations, the relatively low uranium concentration (0.58 to 1.85 $\mu\text{g/L}$) appears due to the reducing condition in groundwater. To ascertain uranium mobilization mechanisms at the low-uranium site, we simulate a single-well push-pull test including injection, drift, and extraction phases using COMSOL-PhreeqcRM coupled reactive transport model. During the injection phase in the model, oxidized surface water is injected to change the groundwater system to an oxidizing condition promoting uranium mobility (insoluble U(IV) oxidized into soluble U(VI)). Based on the results, we investigate the environmental factors influencing the low uranium concentration by analyzing geochemical components and redox conditions. This study demonstrates the potential impact of geochemical environments on uranium transport in natural analogue sites, and further experiments will be conducted to validate the results of reactive transport modeling.

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Keywords: Reactive transport model, Uranium transport, Single-well push-pull test, Natural analogue, Redox condition, COMSOL-PhreeqcRM

Analysis of Fracture Aperture Evolution in Response to Stress Field Changes Using DEM-DFN Model

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In deep geological disposal of high-level radioactive waste (HLW), the natural barrier must physically support the disposal facility and delay the movement of radionuclides for at least hundreds of thousands of years. Therefore, site selection and facility design should be based on a comprehensive understanding of the hydro-mechanical long-term evolution of natural barriers. In Korea, crystalline rock with low permeability is being considered as a potential natural barrier medium. This implies that fractures embedded within crystalline rock mass play a significant role in determining hydro-mechanical behavior, highlighting the importance of fracture apertures due to their direct influence on groundwater flow.

This study aimed to analyze the long-term evolution of fracture apertures using a three-dimensional numerical model. Natural barriers can experience gradual changes in the direction and magnitude of principal stresses over time due to tectonic movements induced by factors such as surface deposition and erosion, glacial environmental changes, and other phenomena. We focused on analyzing the evolution of fracture apertures by considering such stress field changes within long-term scenarios. As an analysis tool, we utilized 3DEC (3-dimensional Distinct Element Code), a widely used distinct element method (DEM) program. Additionally, we constructed a DEM-DFN model using the embedded discrete fracture network (DFN) module. The DFN model was generated to adequately incorporate methodologies proposed in previous studies, including aperture-size relationships, mechanical-hydraulic aperture relationships, and stress-dependent fracture stiffness models. Furthermore, the rock properties and characteristics investigated at the KURT (KAERI Underground Research Tunnel) site were incorporated into the model. Considering the groundwater pressure acting on the fracture surfaces, coupled hydro-mechanical analyses were conducted. We investigated the influence of gradual changes in vertical and horizontal principal stresses on fracture evolution, and the analysis results were presented in relation to the size and orientation of the fractures.

This study presents a block-scale DFN model, serving as fundamental research for the prospective development of an equivalent porous media (EPM) model via equivalent properties. Consequently, it is anticipated to make significant contributions by constructing a site descriptive model for KURT and developing a coupled hydro-mechanical model for demonstration at a generic underground research laboratory (URL).

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Keywords: Natural barrier, Fracture aperture, Long-term evolution, Stress field, Distinct element method, Discrete fracture network

Fault Rock Distribution in Drill Cores for Understanding Deep Brittle Structures Around the KAERI Underground Research Tunnel

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Natural barriers encompassing bedrock and the associated geological structures are characterized by complexity, heterogeneity, invisibility, and unpredictability, necessitating extensive research for performance evaluation. Therefore, it is crucial to investigate site characteristics over an extended period using various methods. In particular, geological structures such as fractures, faults, and joints are significant for the hydraulic properties of surrounding rocks. Among these, fault zones can typically be divided into fault core and damage zone, composed of various types of fault rocks formed under conditions during fault activity. It is essential to note that not all fault rocks serve as conduits for groundwater flow. The maturity of fault rocks, which is related to factors such as clay mineral content, clast abundance, and degree of alteration after events, determines whether fault zones serve as barriers or conduits for groundwater flow. We aim to identify the geometric features and spatiotemporal activity patterns of faults by analyzing the types and distribution of fault rocks in the study area. For this study, we utilized boreholes drilled to a depth of approximately 200 m around the KAERI underground research tunnel. These boreholes were excavated to develop methodologies to understand subsurface lithology and geological structural changes in three dimensions. Selective sampling within the drill cores and optical petrography were used to differentiate between types of fault rocks. In the study area, fault rocks mostly follow fault planes developed in the E-W direction and consist of fault gouge, breccia, and cataclasite. Subsequently, we aim to integrate the deep brittle structures at the site by determining fracture density using WellCAD software and applying criteria for defining structural boundaries derived from previous studies.

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Keywords: Geological structures, Fault zone, Fault rock, Flow barrier, Fracture density

Evaluating Corrosion Characteristics of Cu Between 25-75°C in Sulfide and Chloride Containing Solutions

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Copper (Cu) is considered as a corrosion resistance material suitable for canisters storing High-Level radioactive Waste (HLW) in Deep Geological Repository (DGR). Although Cu is classified as a semi-noble metal, it can be susceptible to corrosion in anaerobic environments due to sulfide ions, which act as strong oxidizing agents. However, there has been limited research to report the characteristics of Cu corrosion at relatively high temperatures (25–75°C), even though the surface of the Cu canister can maintain such temperatures over hundreds of years. This is because it was believed that the period of oxidic corrosion would last for several hundred years. However, recent field tests (Mont Terri FE) have shown that the period of oxidic corrosion diminishes within a few years. Hence, there is a possibility that Cu could corrode under relatively high temperatures in a reducing condition. This study aims to experimentally explore the corrosion behavior of Cu at high temperatures in anaerobic environments by isolating the electrochemical cell with a gas-tight setup. Electrochemical measurements were conducted inside of a glove box to control variables considering DGR conditions, and a heating block was used with precise temperature control. The high purity rolled Cu specimens (> 99.99%) were polished to 0.05 µm. The sulfide solutions (corrosive solution) were prepared by sodium sulfide, sodium chloride, and borate buffer solution which are at least ACS grade. A three-electrode system was employed in the electrochemical cells, with a Potentiostat used for anodic polarization measurements. Results indicate that as temperature rises, corrosion potential decreases, as observed through potentiodynamic polarization curves. Moreover, it is observed that as the chloride concentration increases, the breakdown potential of the passivation or porous film decreases. These results will be utilized as crucial data to prepare for corrosion-related issues to build the DGR in Republic of Korea.

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Keywords: Cu canister, Deep geological repository, High temperature, Corrosion experiments, Potentiodynamic polarization

Solute Transport Visualization in Rough-Walled Rock Fractures Using Microscopic Imaging Technique

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For the safe disposal of high-level radioactive waste, safety evaluations of deep repository facilities are necessary. In Korea, low permeability crystalline rocks are being considered as host rocks for radioactive waste disposal, and fluid flow and solute transport in low permeability rocks mainly occurs through rock fractures. To fundamentally understand the processes governing fluid flow and mass transport in rough-walled rock fractures, research based on experimental observations and measurements is essential. While the micro-Particle Image Velocimetry (micro-PIV) technique have recently been introduced for flow visualization inside microscale fractures, the methodology for mass transport visualization required to analyse and quantify mass transport and dispersion has not yet been established. In this study, we extended the micro-PIV technique to visualize mass transport and measure solute concentrations in rough-walled fractures. Mass transport within rough-walled fractures was successfully visualised using the micro-PIV system, and a calibration process to determine solute concentration from measured fluorescence intensity was presented. The measured concentration was used to determine transport parameters such as solute concentration and dispersion coefficient, demonstrating the transition from macrodispersion to Taylor dispersion-dominated transport. Microscopic imaging analysis revealed that as fluid velocity increases, a higher concentration gradient develops across the fracture aperture, causing rapid solute breakthrough along the main flow channel in the middle of the fracture aperture. Additionally, the study visualized channelized solute transport associated with eddies, accounting for Taylor dispersion and non-Fickian transport for the first time. The improved visualization technique will contribute to experimentally elucidating mass transport processes in rough-walled rock fractures, which have previously been the domain of numerical simulations.

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Keywords: Solute transport, Transport visualization, Rock fracture, Microscopic particle image velocimetry (micro-PIV), Non-fickian, Taylor dispersion

4분과

중·저준위폐기물관리 (ORAL)



Dose Assessment for Residents in Off-Site for New Deployment of Radioactive Waste Incineration Facilities in Korea

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As of March 2024, a total of 126,985 drums of low and intermediate level radioactive waste (LILW) in Korea are stored in nuclear facilities, and about 4,000 drums of radioactive waste are generated annually. According to the “2020 Management Implementation Plan for Low- and Intermediate-Level Radioactive Waste” announced by the Korea Radioactive Waste Agency (KORAD), it is predicted that around 710,000 drums of LILW will be generated by 2095. However, as additional generation are expected depending on operating conditions, such as the construction of new nuclear power plants, the continuous operation and decommissioning of existing plants, the development of technologies such as Small Modular Reactors (SMR), and expansion of radioactive isotope utilization research, the cost burden on operators for the disposal of LILW is expected to increase, and there are concerns about the saturation of disposal facilities with a total storage capacity of 800,000 drums. Therefore, there is a need to introduce a technology with better volume reduction efficiency compared to the currently applied technology in Korea. This is crucial to achieve cost reduction in waste disposal and secure storage space for disposal facilities.

In this study, the largest proportion of LILW was investigated to reduce the amount of radioactive waste disposed of, approximately 45% of the total is identified as combustible waste. Among the volume reduction treatment of combustible waste, incineration technology exhibits the highest volume reduction efficiency with a ratio of 100. For this reason, leading nuclear countries such as France, Japan, and the United States actively utilize incineration technology for radioactive waste. However, there are no radioactive waste incineration facility in Korea, and similar facilities such as vitrification and pyrolysis are either not currently in operation or are operated on a small scale, so utilization of related technologies is limited. The most representative causes for this defensive utilization status are the problem of resident acceptance by gaseous radioactive waste released from incineration facility. Accordingly, in this study, under the assumption that radioactive waste incineration facilities are deployed in Korea, the environmental impact and resident exposure doses of gaseous radioactive effluents generated during the operation of the reduction treatment facility by incineration of combustible radioactive waste were evaluated and compared with the notification standards announced by Nuclear Safety and Security Commission (NSSC), and the safety of the facility’s deployment was analyzed.

Acknowledgements

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Keywords: Radioactive waste, Combustible waste, Incineration, Safety analysis, Dose assessment

Assessment of Structural Stability and Leaching Characteristics of Phosphate-based Geopolymer Waste Form Containing Radioactive Spent Ion Exchange Resins

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Utilizing cementitious waste forms to immobilize radioactive spent ion exchange resins (IERS) is considered inappropriate due to issues such as low waste loading, high leachability of radionuclides, and poor durability. Herein, phosphate-based geopolymers (P-GP) was introduced for the first time to immobilize simulated spent IERS. The 7-day compressive strength of the P-GP waste form decreased as waste loading increased, attributed to the enlargement of pores with higher waste loading. However, the P-GP waste forms containing 40wt% spent IERS met all waste acceptance criteria established by South Korea for compressive strength, thermal cycling, water immersion, and gamma irradiation tests. Although the leaching behaviors of Co, Cs, and Sr differed from those of traditional alkali-activated materials, the leachability index exceeded the prescribed criterion value of 6.0. The leaching mechanism was influenced by a combination of surface wash-off and diffusion processes, or solely diffusion. The P-GP waste form played as a primary physical barrier against the release of radionuclides and displayed minimal structural changes after undergoing waste acceptance criteria tests, indicating its effectiveness in immobilizing spent IERS. These findings provide valuable insights into efficient waste form materials for containing radioactive spent IERS.

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Keywords: Phosphate-based geopolymer, Waste form, Spent ion exchange resin, Leaching, Waste acceptance criteria

A Study on Radioactive Residues From Tin Mining Industry in Indonesia

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The assessment of radiological doses in disposal facilities for NORM (Naturally Occurring Radioactive Material) waste from the tin mining industry holds importance due to the presence of activity concentrations of uranium series, thorium series, and potassium. With Indonesia's tin mining industry producing approximately 165,000 tons of NORM waste annually, mitigating potential harm to the public from radiation becomes important. This study is conducted to compute the total doses from disposal facilities for NORM waste from tin mining, employing the NORMALYSA (NORM And Legacy Site Assessment) code. Through review and analysis of disposal facilities for NORM waste from tin mining industries, various scenarios and pathways were investigated. These facilities typically contain mine tailings and final slag with relatively low-level concentrations of radionuclides. Parameters utilized in the NORMALYSA code are derived from previous studies concerning NORM waste management in Indonesia. By simulating the pathways, putting parameter values, using the NORMALYSA code, the study aims to analyze the contributions of various factors to the total doses incurred. This result approach is recommended to strengthen safety measures surrounding disposal facilities for NORM waste. The findings of this research are composed to provide valuable insights and recommendations for regulatory bodies, stakeholders within the tin mining industry, and local government authorities. By recognizing and addressing the potential radiological impact, stakeholders can implement measures to mitigate risks effectively. Ultimately, this study strives to contribute to the enhancement of safety for the public and the sustainable management of NORM waste in the context of Indonesia's tin mining industry.

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Keywords: NORM, Disposal, Tin mining, NORMALYSA, Total doses

Study for the Slope Stability of Engineering Barrier on Shallow Radioactive Waste Disposal Facility

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2nd phase surface radioactive waste disposal facility is under construction at Low and Intermediate Level Radioactive Waste (LILW) disposal site in Gyeongju. It comprises with three barriers such as 20 sets of concrete vaults placed on excavated ground, backfills and artificial slope with engineering barriers. Its slope is firstly introduced facility in Korea and thus have some uncertainties in terms of materials, design and execution. Thus, pilot slope is built up in LILW site for further study.

Pilot slope is composed of multilayers, monitoring equipment installed inside and on them. Its monitoring started on Oct 31, 2022 and accumulated monitoring data is analyzed. In addition, slope stability analysis is performed for establishing contingency plan to prepare for the management of future 2nd phase surface radioactive waste disposal facility. Local design code is applied and seismic analysis is also included. Based on the analysis, simplified management criteria is suggested.

Finally, Pilot slope management plan is shown under the experience of its execution and monitoring.

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Keywords: Engineering barrier, Radioactive waste disposal facility, Slope stability

The Worker's External Dose at Maintenance Work Due to Failure During Operation of the Spent Resin Mixed Waste Treatment Facility From Wolseong NPP Unit 1

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Worker's dose was evaluated based on the maintenance scenario at equipment failure of the spent resin mixed waste treatment facility. Radioactive spent resin mixed waste is generated during the operation of the heavy water reactor at Wolseong Nuclear Power Plant Unit 1. While the waste treatment facility is operated, the radiological safety of worker must be ensured in preparation for normal and emergency situations. Therefore, worker's dose should be assessed according to the worker's maintenance scenario based on the failure of each equipment of the spent resin mixed waste treatment facility to confirm the radiation safety of workers conducting maintenance work of the failure.

The radioactivity concentration of nuclides ^{60}Co , ^{137}Cs , ^{134}Cs , ^{95}Nb and ^{125}Sb which were detected in the mixed waste including spent resin, spent activated carbon and spent zeolite from Wolseong NPP Unit 1, was considered for estimation of the worker's dose.

VISIPLAN code was used to evaluate scenario-based external dose to the worker, where the equipment of the mixed waste treatment facility subject to failure included a transfer/injection container, a waste liquid tank, a separation device, a spent resin storage tank and a mixed waste storage tank. In the event of equipment failure, work time and the number of workers were pre-determined considering the size of the equipment, the repairing time consumed to maintenance and the expected level of radioactive contamination, where 40 hours of working time and 2 workers were expected for decontamination and disassembly/cutting each and 24 hours of working time and 4 workers for the removal of pipes.

The worker's dose was evaluated to range from 7.6×10^{-4} mSv to 1.24 mSv for decontamination and disassembly/cutting each and 1.368×10^{-4} mSv to 0.22 mSv for the removal of pipes. Accordingly, it was seen to meet the worker's dose limit in terms of external dose and would be followed by further evaluation considering the internal dose to confirm the radiological safety of the worker.

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Keywords: Treatment facility, Maintenance scenario, External dose, VISIPAN4.0

PSR (Periodic Safety Review) Plan for Operating Gyeong-ju LILW Disposal Facilities

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Radioactive waste disposal facility is required to demonstrate their safety by conducting comprehensive evaluations every 10 years from the date they begin operations. The low-and intermediate level radioactive waste (LILW) disposal facility in Gyeong-ju, the first of its kind for defense waste in Korea, is therefore obligated to perform the country's inaugural periodic safety evaluation.

According to the law, the medium and low-level disposal facility in Gyeong-ju, which started operations in 2014, must undergo an evaluation for a 10-year period after its opening and submit an evaluation report to the Nuclear Safety and Security Commission. The Korea Nuclear Environment Corporation is in the process of preparing for this periodic safety evaluation of the disposal facilities. The plan for the periodic safety evaluation includes collecting and analyzing data in 2024, submitting a periodic safety evaluation report to the Nuclear Safety and Security Commission in 2025, and subsequently undergoing a review by the regulatory agency.

Per Article 104-3 of the Enforcement Decree of the Nuclear Safety Act, the periodic safety evaluation will encompass an assessment of 12 factors. These factors are integral to the comprehensive evaluation of the safety of the disposal facilities.

Acknowledgements

This work was supported by the Korea Radioactive Waste Agency (KORAD).

Keywords: LILW, Disposal facility, PSR

Non-Destructive X-ray Computed Tomography Characterization of Novel Wasteforms Targeting ^{14}C Solidification: Results of a Joint ROK-UK Project

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Metakaolin geopolymers have shown promise for ^{14}C immobilization and final disposal. However, quantitative internal structure characterization of wasteforms remains elusive without destruction X-ray computed tomography (CT) can enable non-destructive evaluation (NDE) allowing for internal structural imperfections to be identified and quantified as part of defect metrology. Therefore, it was proposed to qualitatively and quantitatively analyse the internal structure of fabricated geopolymer wasteforms using X-ray CT NDE for formulation validation, deformity detection, and failure prediction.

To accurately quantify collected X-ray CT data a spatial resolution target (image quality indicator, IQI) can be used. An IQI incorporates features of known dimensions that can be measured accurately and verified independent of the X-ray CT system, in effect enabling calibration of the collected X-ray CT data. Use of an IQI in this way allows for accurate determination of the spatial resolution, detection limit and quantification of the measurement error, and thus determination of measurement confidence.

In this study, a bespoke X-ray CT spatial resolution target (IQI) has been developed and 3D printed, enabling defect metrology of geopolymer wasteforms. Geopolymer specimens made of Sodium Alumino-Silicate Hydrate (N-A-S-H) or Potassium Alumino-Silicate Hydrate (K-A-S-H) pastes were prepared to which carbonates (CaCO_3 or SrCO_3) were loaded as representative wastes at either 20, 40, or 60wt%. The geopolymer pastes were cast in cylindrical molds to form miniature wasteforms with a 23 mm inner diameter, 46 mm height, and a volume of 19.1 cm^3 . After curing, select specimens were exposed to a compressive strength load of 3.44 MPa, the minimum waste acceptance criteria (WAC) for the Korean disposal facility, and leaching tests. The wasteform specimens were scanned using a V-tome-x M300 (Waygate Technologies) at both low resolution (LR) and high resolution (HR), visualized, and assessed for fractures and defects as part of formulation validation and deformity detection.

The IQI was successfully manufactured and used for determining measurement confidence. Detection limits of $68.96 \mu\text{m}$ (LR) and $27.82 \mu\text{m}$ (HR), approx. $\times 1.82 - \times 1.85$ effective pixel size, were achieved. The geopolymers were found to form consistently and homogeneously with only minor porosity (Avg = 0.8%). Compression test results indicate that all samples would pass the WAC of 3.44 MPa and that compression testing is not fatal.

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Keywords: X-ray computed tomography, Scan spatial resolution, Metrology, Non-destructive evaluation, Wasteform, Resolution limit measurement confidence

Mechanistic Investigation of Molybdenum Immobilized in Geopolymer Matrices

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This study investigates the loading capacity and behavior of molybdenum species within geopolymer matrices. Geopolymers (GP), as environmentally friendly alternatives to traditional cementitious materials, have attracted significant attention for their potential applications in various fields, including radioactive waste solidification and environmental remediation. Molybdenum (Mo) in HLW raffinates originating from the reprocessing of spent nuclear fuel is usually immobilized by the vitrification process. Higher concentrations of Mo tend to make highly soluble crystalline molybdate phases. Therefore, understanding and interaction of Mo species with GP matrices is crucial for optimizing the immobilization efficiency and ensuring the long-term stability of immobilized waste forms.

We have synthesized a series of Mo-hosted GP solid matrices spanning a range of Mo concentrations from 0 to 8wt%. These matrices underwent product consistency tests (PCT) according to ASTM standards, revealing exceptional Mo retention rates of up to 90% within the geopolymer (GP) matrix. X-ray diffraction (XRD) patterns unveiled sodium molybdate ($\text{Na}_2\text{Mo(VI)O}_4$) as the predominant phase with increasing Mo content and thermogravimetric analysis (TGA) showcased a remarkable thermal stability without any degradation products except the removal of physisorbed and interlayer bound water. X-ray photoelectron spectroscopy (XPS) corroborated the presence of Mo(VI) species across all samples (2–8wt% Mo). Scanning electron microscopy (SEM) and high-resolution transmission electron microscopy (TEM) mapping depicted an amorphous and porous structure with uniform Mo distribution before and after leaching tests. These insights shed light on the mechanisms underlying molybdenum immobilization, offering valuable guidance for developing effective geopolymer-based waste containment systems for managing radioactive waste.

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Keywords: Molybdenum, Solidification, Geopolymers, Radioactive waste management

The Study on Optimal Solidification Conditions According to Composition Ratio of the Corrosion Products in the Sludge Waste From Nuclear Power Plant

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This research confirmed the optimal solidification conditions such as heating temperature and mixing ratio of additives according to the mixing ratio of each component in the sludge waste. Most of metallic structures in nuclear power plant, consist of stainless steel. Generally, stainless-steel has a high corrosion resistance. However, when these metallic structures exposed to various radioactive fluid waste based on water and moisture for a long time, corrosion product by oxidation can be generated. Since main components of stainless steel are iron (Fe), nickel (Ni), and chrome (Cr), corrosion products generated from stainless steel are oxidation materials based on Fe, Ni, and Cr. These corrosion products can be separated from metallic structures and accumulated as radioactive sludge waste, which, in Korea, should be solidified satisfying the disposal acceptance criteria for its disposal. The representative oxides of stainless steel (Fe_2O_3 , NiO , Cr_2O_3) have very high melting temperatures of 1,565, 1,955 and 2,435°C, respectively, making it difficult to melt the sludge waste based on such oxides. Instead, the sludge waste can be solidified in the relatively low temperature by using various additives (glass, cement, asphalt, polymer, etc) without the direct melting it. In this study, the glassy material of ferro frit powder, which had the different softening temperature (glass transition temperature) according to its various components' ratio, was used as additives for solidification of sludge waste. Ferro frit powder has such components as Na_2O , K_2O , CaO , Al_2O_3 , B_2O_3 , SiO_2 and ZnO , where ferro frit 3195 used in this study had the softening temperature range of 816–927°C. The melting point of the sludge waste mixture can be lowered bringing the stable solidification if the mixing ratio of ferro frit 3195 to sludge waste is high whereas it can increase the volume of solidified sludge waste. Especially, the more mixing ratio of the corrosion product as Cr_2O_3 with higher melting point, the more difficult stable solidification. Adding 10% of ferro frit 3195, it was confirmed that oxide sludge ($\text{Fe}_2\text{O}_3\text{:NiO:Cr}_2\text{O}_3 = 70\text{:}18\text{:}12$) heated at 950°C was stable. When the mixing ratio of $\text{Fe}_2\text{O}_3\text{:NiO:Cr}_2\text{O}_3$ changed from 70:18:12 to 50:28:22 and 30:38:32, although the additive ratio of ferro frit 3195 increased from 10 to 25%, stable solidified sludge cannot be formed at 950°C. However, it was confirmed that solidified sludge satisfied leachability index higher than 6, compressive strength higher than 3.44 MPa at the temperature higher than 1,050°C.

Keywords: Corrosion product, Radioactive sludge, Ferro frit, Leachability index

Sampling Method Comparison for Homogeneity Assessment of Cementitious Radioactive Waste During Compressive Strength Measurement

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Low- and intermediate-level waste (LILW) should be solidified before disposed of in the repository to remove the fluidity of the waste. In Gyeongju LILW repository, the solidified waste can be disposed of in the repository after satisfying waste acceptance criteria (WAC). The criterion directly related to the structural stability of the waste is compressive strength criterion. The compressive strength of the specimen cored from the waste is measured, and the waste is judged to be structurally stable if the average strength satisfies the criterion. However, in addition to the compressive strength, from the Nuclear Safety and Security Commission (NSSC) notice, the waste with fluidity should have homogeneity.

The compressive strength of cementitious material is generally high enough to maintain its long-term structural stability. However, the compressive strength of cementitious material generally follows the compressive strength of weakest part. Then, the cement solidified waste without homogeneity may have the weaker compressive strength than the expected strength. In addition, the inhomogeneous part of the solidified waste can cause the locally high leaching rate of radionuclides inside the waste. Therefore, the homogeneity is critical factor for the stability of the waste inside the repository that should be assessed.

However, a method for assessing the homogeneity of the solidified waste is not clearly developed. Generally, the homogeneity of the solidified waste is assessed by measuring the compressive strength of the specimen from the cores of solidified waste drum. The compressive strengths of the specimens are measured, and the waste is judged to be homogeneous if the deviation of measured compressive strength is low enough. Nonetheless, the homogeneity assessment result can be different according to the number of specimen and coring positions due to random position of inhomogeneous region of the waste. Thus, the development of homogeneity assessment guideline for solidified radioactive waste should be performed.

In this study, several sampling methods are adopted to derive the adequate coring position with the highest efficiency. In addition, the direction of increasing the number of specimens showing a higher detection rate of inhomogeneous waste is compared. The detection rate of inhomogeneous solidified waste is calculated by modeling the inhomogeneous waste with random position of inhomogeneous region through MATLAB. As a result, the most efficient sampling method and the direction of increasing the number of specimens are obtained. The result of this study is expected to become a steppingstone for the development of homogeneity assessment guideline.

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Keywords: Cementitious material, Homogeneity assessment, Modeling, Radioactive waste, Waste acceptance criteria

Homogeneity Evaluation Test of Solid Waste Form Containing Spent Ion Exchange Resins Using Statistical Analysis

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The homogeneity of waste is important in solidifying radioactive waste because heterogeneity of the solid waste form can adversely affect characteristics of solid waste form. Despite the importance of homogeneous solidification, there is no established method to evaluate the homogeneity of solid waste form as waste form acceptance criterion. In this study, we compared the compressive strength and leaching characteristics of homogeneous samples and artificially fabricated heterogeneous solid waste form samples to investigate the appropriate method for determining homogeneity using statistical analysis and laser-induced breakdown spectroscopy (LIBS). The samples were fabricated by solidifying simulated spent ion exchange resins (IERS) adsorbed non-radioactive Cs with a metakaolin-based geopolymer, and a cylindrical mold (2.9 cm in diameter, 6.0 cm in height) was used. The geopolymer was synthesized by mixing metakaolin and K_2SiO_4 solution (alkali activator) in 1:1.2 mass ratio. The total waste loading of the samples was fixed at 15wt%, and the heterogeneous samples were fabricated by varying the waste loading vertically in one mold. Three types of heterogeneous samples with different waste loading at top and bottom sections like 10+20wt%, 5+25wt%, and 0+30wt%, respectively were fabricated. For the compressive strength test, only the 0+30wt% sample showed slightly lower compressive strength. The leaching test also showed a similar trend to the compressive strength test, confirming that only the 0+30wt% sample showed higher leaching rate than the other samples. The sample was divided into three parts and Cs intensities were measured using LIBS on surfaces of parts. The normality test was performed on the measurement distributions using Shapiro-Wilk, and Kolmogorov-Smirnov tests. The test results confirmed that measurement distribution from homogeneous sample was closer to normal distribution. Since the distribution measured from the homogeneous samples will have low variance, the coefficient of variances (CoVs), which can indicate the dispersion of the distribution, were also calculated. CoVs increased in proportion to the heterogeneity of the sample. The normality test and CoV calculation showed that the 5+25wt% and 0+30wt% samples have significant heterogeneity. The homogeneity of solid waste form can be determined by statistical methods conducted in this study, but there can be difference between the heterogeneity level judged by statistical methods and the level that can affect performance. Therefore, the homogeneity of the solid waste form should be evaluated considering performance indicators as well.

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Keywords: Homogeneity of solid waste form, Spent ion exchange resin, Geopolymer, Normality test, Laser-induced breakdown spectroscopy

Effect of Si/Al Ratio in Metakaolin-based Geopolymer Formulation on Immobilization of Radioactive Selenium

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Selenium, present in trace amounts in spent nuclear fuel, is regarded as a high-priority concern due to its long half-life (approximately 3.27×10^5 years) and high solubility due to forming soluble oxyanions: selenite (IV) and selenate (VI). Selenite forms strong inner-sphere complexes with sorbents, leading to lower mobility, whereas selenate forms weak surface complexes with lower sorption affinity on the minerals, resulting in higher mobility. In addition, ettringite has been studied for incorporating selenate through the coprecipitation process, resulting in selenate ettringite, $\text{Ca}_6\text{Al}_2(\text{SeO}_4)_3(\text{OH})_{12} \cdot 26\text{H}_2\text{O}$. However, selenate ettringite still needs binding materials to be immobilized like waste form to prevent potential selenate release into the environment. Ordinary Portland cement, used for a typical low-temperature waste form contains sulfate which competes with selenate for ettringite. This could lead to the disintegration of selenate ettringite into nanoparticles. In this study, metakaolin-based geopolymer was used as a waste form to immobilize selenate ettringite. We investigated the effects of Si/Al molar ratio on the Se leaching behavior and the mechanical properties of the geopolymer waste form containing selenate ettringite. The Si/Al molar ratio of geopolymers was adjusted to 2.0, 1.75, and 1.5, respectively. The 10wt% selenate ettringite was loaded in each geopolymer sample. The geopolymer pastes with different Si/Al molar ratios were poured into a cylindrical mold (diameter: 2.9 cm and height: 6.0 cm), then cured at room temperature for 1 day and 60°C for 6 days. The X-ray diffraction (XRD) analysis, Fourier transform infrared (FTIR) spectroscopy, field-emission scanning electron microscope (FE-SEM), energy-dispersive spectrometer (EDS), solid-state ^{29}Si magic-angle spinning (MAS) nuclear magnetic resonance (NMR) spectroscopy, 7-day compressive strength, and leaching tests were performed to evaluate the properties of geopolymer with different Si/Al molar ratios containing selenate ettringite. The compressive strength was not proportional to Si/Al molar ratio, and the lowest compressive strength (10.61 ± 0.02) was achieved in geopolymer waste form with a Si/Al molar ratio of 2.0. After the 5-day leaching test, the cumulative fraction leached (CFL) decreased from 0.97 to 0.84 and the leachability index slightly increased as the Si/Al molar ratio of geopolymer decreased. The microstructure characteristics indicated that as Si/Al molar ratio increases, selenate ettringite becomes disintegrated. These results imply that to effectively immobilize selenate ettringite, a waste form with a low Si/Al molar ratio is a promising candidate.

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Keywords: Selenium immobilization, Selenate ettringite, Metakaolin-based geopolymer, Si/Al ratio

Analysis of Solidified Paraffin Characteristics From Nuclear Power Plants and Consideration of Disposal Plans

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Radioactive liquid waste produced during the operation of domestic nuclear power plants is processed through the Liquid Radwaste System (LRS). Before the introduction of Korean standard nuclear power plants, the Concentrated Water Dry System (CWDS) was utilized to evaporate this waste and convert it into a powdered form. From 1995 to 2012, this powdered waste was solidified using paraffin and stored within the nuclear power plant. Currently, approximately 3,637 drums (each with a 200 L capacity) of paraffin-solidified waste are stored across the Gori (987 drums), Hanbit (1,405 drums), and Hanul (1,245 drums) nuclear power plants. The permanent disposal of this paraffin-solidified waste poses significant challenges due to unsatisfactory homogeneity and leaching characteristics, necessitating the development of an optimal disposal strategy.

This study delves into the physical properties of paraffin, the chemical properties of boron-infused radioactive waste dry powder, and the physicochemical properties of paraffin-solidified powder. It proposes suitable alternative technology for the treatment of drums containing boron-infused radioactive waste powder. The process involves efficiently extracting the solidified body from the drum, ensuring complete separation of the paraffin from the waste. The separated paraffin must then be disposed of in a manner that conforms to the criteria for self-disposal, while the radioactive waste is handled according to the repository's acceptance criteria.

In this study, an examination of paraffin, the solidified body, and the crushed solidified particles was performed. And various disposal methods, including layer separation technique for paraffin removal, a PC-HIC (Poly Concrete-High Integrity Container) charging method following the crushing of the solidified body, and a miniaturized HIC overpacking approach were compared and reviewed.

Keywords: Liquid radwaste, Paraffin, Solidification, Boron-radioactive

Fabrication and Durability Evaluation of Cesium Pollucite Waste Form

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Many fission products are present in spent nuclear fuel (SNF). In the inventory of nuclides in SNF, cesium and strontium are known to be the largest contributors to the decay heat of SNF. Given the decay heat and the disposal area of the repository, it is advantageous to separate these radionuclides from SNF to reduce the disposal burden.

Radioactive cesium is highly mobile in aqueous media, due to its high solubility in water. Therefore, when selecting a matrix for immobilization, thermal stability should be considered along with ensuring the chemical durability of the cesium in the disposal environment. Thermal stability is more advantageous in ceramic matrix than in glass matrix. Furthermore, Cs pollucite has been reported to have excellent chemical durability due to the aluminosilicate framework that immobilizes cesium ions stably.

In this study, we prepared waste forms aimed to dispose cesium-containing aluminosilicate waste, and performed characterization and centerline temperature evaluation. The simulated wastes were prepared by using Cs_2CO_3 as cesium precursor and mixing heat-treated kaolinite composed of mullite and SiO_2 in mass fractions of 0.1, 0.2, 0.3, and 0.4. Subsequently, each sample was calcined at 1,200°C for 2 hours. The prepared samples were ground and mixed with 2.0wt% Polyvinyl alcohol as a binder, dried and re-milled to a fine powder. The powder was fabricated into cylindrical pellets with a diameter of 25 mm at different pressures (5 or 10 MPa) and sintering temperatures (1,400 or 1,500°C) using the cold pressing method. The characteristics of the resulting pellets were evaluated by XRD, SEM, EDS, PCT-A and calculation of the centerline temperature.

The results of this study can provide data on the proper waste loading, fabrication conditions, and centerline temperature of the waste form for immobilizing cesium.

Keywords: Radioactive cesium, Ceramic waste form, Characterization, Chemical durability, Centerline temperature

Ceramic and Cermet Waste Forms for Immobilization of Radioactive Sr Nuclide

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The Korea Atomic Energy Research Institute (KAERI) is currently developing the waste burden minimization technology to reduce the environmental burden caused by the disposal of spent nuclear fuel and maximize the utilization of the repository facility. The technology incorporates a nuclide management process that can maximize disposal efficiency by selectively separating and collecting major nuclides in spent nuclear fuel. In addition, for efficient storage facility utilization, the short-term decay heat generated by spent nuclear fuel must be removed from the waste stream. To minimize the short-term thermal load on the repository, it is necessary to separate heat-generating nuclides. Sr nuclides, especially Sr-90, must be separated because it generates significant amounts of thermal energy during radioactive decay. Therefore, KAERI has developed a technology for separating Sr nuclides from Group II nuclides separated through the nuclide management process, and these separated nuclides should prepare as stable waste forms for interim storage and disposal. In this study, Sr ceramic waste form, SrTiO_3 , was prepared by the solid-state reaction method for immobilization of separated Sr nuclides and evaluated the physicochemical properties. We also evaluated the radiological and thermal characteristics of the Sr waste form, predicting the composition of Sr nuclides separated from spent nuclear fuel. Furthermore, we introduced the ceramic metal (cermet) waste form for thermal management of high heat generating ceramic waste form and evaluated the physicochemical properties of these waste form. These results provided fundamental data for interim storage and management of Sr waste from.

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

4분과

중·저준위폐기물관리 (POSTER)



Ultrasonic Decontamination Process to Treat Actual Uranium-Contaminated Gravel Under Neutral pH Condition

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This research focused on developing a thorough approach to decontaminating soil and gravel contaminated with uranium (U), considering both economic and practical aspects. The main aim was to assess the efficacy of using ultrasonic technology for decontaminating soil and gravel affected by U. Various experiments were conducted to analyze the distribution of U in the contaminated samples, including soluble U at neutral pH, fine insoluble U particles on the surface, and U chemically bound within the material. Following these experiments, a method utilizing ultrasonic treatment was applied to decontaminate crushed gravel contaminated with U, aiming to meet the clearance level guidelines ($< 1 \text{ Bq} \cdot \text{g}^{-1}$) set by the International Atomic Energy Agency. Results showed that the effectiveness of decontamination generally increased with higher ultrasonic frequencies, with samples containing lower U concentrations meeting clearance levels after treatment with higher frequency ultrasonic waves under neutral pH conditions. Furthermore, the presence of $\text{OH}^\cdot$ in solution during ultrasonication was found to facilitate the removal of chemically bound U from the gravel. Finally, through an economic analysis comparing the proposed approach with previous studies and with the permanent disposal of contaminated materials without decontamination, it was concluded that the suggested comprehensive zero liquid discharge system represents a feasible and practical strategy for managing U-contaminated soil and gravel.

Acknowledgements

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Keywords: Uranium, Contamination, Ultrasonication, Neutral pH

Direction for the Development of Thermal Treatment Technology for Intermediate- and Low-Level Radioactive Waste (ILRW)

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Thermal treatment of ILRW, which occupies a large volume, is rational from safety and economic perspectives, as it allows for significant volume reduction and stabilization to meet disposal criteria. Recently, KHNP has included drum processing in its master plan for ILRW treatment and disposal. However, there is a need for treatment technologies, particularly for waste such as cellulose waste, which generates Isosaccharinic acid (ISA) in past disposal environments and does not satisfy disposal conditions. In such cases, thermal treatment is considered a highly suitable method. However, the application of thermal treatment technology for domestic ILRW, such as the KAERI “RI Waste Incineration Facility” and the “Vitrification Facility” at the Hanul #3, has been suspended. Several considerations need to be addressed in the development of thermal treatment technology for ILRW:

- It is not desirable to solve various waste issues simultaneously with a single process technology: Processing diverse waste types such as paper, fibers, rubber, wood, PVC, and vinyl in a single process leads to problems such as premature degradation of process refractory materials due to excessively high process temperatures, increased secondary waste generation, complex processing steps, facility size expansion, and compromised economics.
- Measures need to be in place for the treatment of acidic gases present in off-gases: Preventive measures are required to address corrosion of process piping and the generation of dioxins due to the presence of chlorine (Cl) in waste, as well as to prevent thermal NO_x generation caused by air in-leakage during process vessel sealing. Fortunately, since 2004, radioactive waste has been well segregated by type at nuclear power plants. However, there is a need for more detailed separation and self-restraint in the use of PVC material at NPPs.
- Technological objectives for thermal treatment processes need to be adjusted reasonably: Preventive measures should be simplified to satisfy disposal suitability levels based on the principle of disposing of residual materials (ash, scrubber bottom salt etc.) as a uniform solid mass. The throughput of thermal processes should be viewed from the perspective of disposal of final products rather than waste input, and disposal suitability should be determined based on online sampling and analysis using Process Control Program (PCP).

Fortunately, through the development of thermal treatment technology for ILRW, many key technological elements have been advanced, and valuable lessons have been learned. Furthermore, since the physical and chemical characteristics of urban and industrial waste are not significantly different from radioactive waste, most related technologies can be absorbed and utilized in the process of reducing volume and stabilizing waste for separation into residential areas. European and Japanese countries have long been reducing the volume of ILRW through thermal treatment. Therefore, the development of thermal treatment technology for domestic waste is urgently needed to secure competitiveness in nuclear power technology and decommissioning technology exports.

Keywords: Thermal treatment, ILRW, Disposal criteria, Off-gas, PCP

Preliminary Study on the Uranium Precipitation Method for Wastewater Treatment From Soil Washing Processes Containing Uranium Isotope

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Radioactively contaminated soil waste possibly occurred due to the decommissioning of nuclear facilities, accidents, etc. At KAERI, a large amount of uranium-contaminated soil waste was stored in around 8,000 drums (200 L per drum), originating from the decommissioning of the uranium conversion facility. The construction of the radioactive waste disposal site has been completed, so it is necessary to dispose of the radioactive waste quickly to ensure public safety. Generally, radioactive waste should be disposed of at a permanent disposal site, but the high regulatory standards and significant disposal costs pose critical challenges. Additionally, the contamination characteristics of the soil indicate surface contamination caused by uranium only, which can reduce the volume of radioactive waste easily by decontamination rather than different types of radioactive waste. Recently, we have developed a treatment process for the volume reduction of soil waste. This process leaches uranium from the soil waste using an acidic chemical solution, effectively reducing uranium to below the clearance criteria. However, radioactive wastewater containing uranium was generated during the process, which poses a critical issue for the volume reduction of radioactive waste.

In this study, we conducted a preliminary investigation into the uranium precipitation method for treating wastewater from soil-washing processes containing uranium isotopes. The experiments proceeded using simulated soil-washing wastewater. General soil was washed with 0.5 M H_2SO_4 , followed by the dissolution of the uranium reagent in the solution. We compared the effectiveness of different chemical precipitation methods by evaluating their uranium removal efficiency. This helped us identify the most suitable method for treating this complex wastewater. As a result, uranium could be effectively removed from the wastewater to levels below the discharge limit ($< 1 \text{ mg L}^{-1}$). Therefore, the secondary wastewater could be effectively reduced using the precipitation method, thereby enhancing the efficiency of the treatment process.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 522230-24).

Keywords: Wastewater treatment, Precipitation, Uranium separation, Soil decontamination

K-DWaIT: An Inventory Verification Tool for Decommissioning Waste Using Microsoft Excel and Visual Basic Add-in

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After the permanent shutdown of Kori Unit 1 in 2017, the decommissioning of nuclear power plants (NPPs) has emerged as a significant issue. In particular, during the decommissioning of NPPs, a large quantity of decommissioning waste with various levels of radioactivity is generated across multiple areas in a short period. Therefore, accurately managing the physical inventory (volume, mass, etc.) of all generated decommissioning waste is essential for the safety of waste management during decommissioning. Furthermore, to secure the trust of the public and stakeholders regarding the safety of decommissioning, it is important to prevent the loss of decommissioning waste or inventory records.

In this regard, IAEA General Safety Requirements (GSR) Part 5 defines the concept of traceability, and GSR Part 6 recommends that traceability is ensured for all waste generated during decommissioning. Specific Safety Guides (SSG) No. 40 recommends the development and implementation of a tracking system for waste packages and SSG No. 47 recommends maintaining records of key decommissioning actions during decommissioning. Although the records of decommissioning waste over time during the decommissioning process should be assessed and demonstrated for adequacy by the licensee, regulatory body also need independent assessments to verify the adequacy of the assessment and management of decommissioning waste inventory by licensee.

Therefore, K-DWaIT (Kyung Hee - Decommissioning Waste Inventory verification Tool) has been developed to manage waste generated at decommissioning sites by dividing it into three characteristics: mass, radioactivity, and item type, and to pre-verify abnormal inventory changes that may occur during the pre-disposal process from the stance of regulatory body.

Acknowledgements

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Keywords: Decommissioning waste, Inventory verification, Material balance, Code, K-DWaIT

Comparative Study on Safety Standard History of IAEA Including GSG-18 for Clearance

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According to IAEA PRIS, the number of nuclear power plants scheduled for decommissioning is increasing, which will consequently lead to an increase in radioactive waste generated. A significant portion of the radioactive waste produced during the decommissioning process of nuclear power plants consists of very low-level radioactive waste that does not require radiation safety regulation. This concept is commonly referred to internationally as “Clearance”, and the IAEA defines clearance as the removal of regulatory control by the regulatory body from radioactive material or radioactive objects within notified or authorized facilities and activities. Clearance is recommended by many international organizations and countries for the efficient management of large quantities of very low-level radioactive waste. The IAEA has officially defined the concept of clearance and clearance levels by publishing RS-G-1.7, GSR Part3, and GSG-18, which many member countries have incorporated into their own legislation and regulations.

In 1996, the concept of clearance was mentioned in IAEA Safety Series 115; however, specific criteria such as clearance levels for radionuclides were not provided. It wasn't until 2004 that RS-G-1.7 was officially published, defining the concept of clearance and providing clearance levels for radionuclides. After the publication of RS-G-1.7, the IAEA released GSR Part 3 in 2014, followed by GSG-18 in 2023. GSG-18 incorporates the content of the previously published RS-G-1.7 and GSR Part 3, but detailed guidance has been added. It incorporates dose criteria for clearance, such as 10 μ Sv annually in realistic scenarios and 1 mSv annually in low probability scenarios. Furthermore, it introduces new content on clearance for liquid and gaseous materials, as well as specific clearance. As member countries typically incorporate guidance published by the IAEA into their own legislation and regulations, it is expected that many countries will update the implementation of their clearance systems in accordance with the new content presented in GSG-18.

This study conducted an analysis on history of IAEA including GSG-18 for clearance. It is anticipated that this study can be utilized to analyze future developments in clearance systems by international organizations and individual countries.

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Keywords: Clearance, IAEA, RS-G 1.7, GSR part3, GSG-18

Development of Sample Handling and Transportation Processes for the Analysis of Radioactive Waste Characteristics in Hot Cell Facility

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According to Korea Hydro & Nuclear Power's announcement, it is expected that more than 80,000 drums of radioactive waste will be generated during the decommissioning process of Kori Unit 1. For the acceptance inspection by the Korea Radioactive Waste Agency, it is essential to determine physical, chemical, and radiological characteristics of the waste. therefore, comprehensive analysis of the waste characteristics is crucial. However, there is a shortage of facilities and personnel for this purpose, and even with the participation of private companies, there are limitations on the amount that can be processed.

To enhance decommissioning technology and meet the rapidly increasing demand for the analysis of radioactive waste characteristics, the Korea Research Institute of Decommissioning (KRID) was established in 2020. KRID is working to alleviate bottlenecks in analysis resulting from the decommissioning process. This effort includes establishing facilities such as hot cells for analyzing intermediate-level waste and providing expert training.

The handling of decommissioning waste requires not only appropriate facilities and trained personnel but also efficient and safe procedures. Particularly concerning intermediate-level waste with high level radiation, meticulous care must be taken throughout processes such as sampling, transportation, packaging, and analysis. Therefore, it is imperative to establish an overarching procedure that considers these intricacies.

In this paper, the entire process that should be considered from the sampling of decommissioning waste to its transportation to the hot cell in KRID is analyzed and systematized, and consequently, the following basic procedure has been developed.

1. In the sampling process, selecting sampling targets, locations, and determining sizes and numbers of samples are done to ensure that they adequately represent various waste generated from cutting and decontamination processes.
2. For the packaging and transportation process, design of packaging and handling devices considering the intensity of the collected waste and setting transportation plans and routes considering scenarios of exposure to drivers and residents during transportation.
3. In the hot cell facility, inspection of waste transport containers and transfer to hot cells. After opening the samples in a fully sealed hot cell, they are transferred to an analysis cell, and the transport containers are decontaminated for reuse.
4. Processing the samples to appropriate sizes for each analysis purpose or preprocessing them into suitable forms.
5. Performing radioactive analysis on the processed samples and conducting physical tests (tensile, fatigue, hardness, impact) and microstructural analysis if necessary.

Refining the methodologies for sampling, transportation, packaging, and analysis, particularly for intermediate-level waste, future research will focus on optimizing this basic process by enhancing the efficiency and safety of waste handling procedures.

Acknowledgements

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Keywords: Decommissioning, Intermediate level waste, Hot cell, Transportation, Package, Analysis

Radiochemical Decomposition of Oxalic Acid in Radioactive Liquid Waste

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Oxalic acid is one of the proposed intermediate products in the multistage phenol degradation into CO₂ under the ionizing irradiation of aqueous solution. On the other hand, radiation-chemical conversion of liquid radioactive waste significantly predefines their safe storage. Consequently, detailed knowledge of the processes interlining the radiolysis of oxalic acid which is one of the components of liquid radioactive waste remains still of interest. Co-60 gamma ray radiolysis of aqueous oxalic acid solution (1×10^{-2} M) was investigated within the dose range up to 80 kGy. The value of pH increases from 2.0 to 4.0 and approximately 90% of oxalic acid and its derivatives are degraded at the dose of 80 kGy. Dissolved O₂ affects to the degradation of oxalic acid at the initial stage due to total consumption of O₂ at the dose of 2 kGy. Formation of CO₂ occurs at a higher rate at the initial stage until the dissolved O₂ is fully consumed. The kinetic model of oxalic acid degradation in aqueous solution under the gamma irradiation was tested. The suggested mechanism complies with the experimental data both of our own and of that reported earlier. Approximately 90% oxalic acid and its derivatives are degraded at the dose of 80 kGy. Significant increase in pH under gamma irradiation occurs.

Acknowledgements

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Keywords: Oxalic acid, γ -irradiation, Kinetic model, Radiation-chemical yield

Understanding Other Laws for Clearance Level Radioactive Waste Focused on Chemically Decontaminated Uranium Contaminated Soil Waste

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Based on the “Regulations on Radioactive Waste Classification and Self-disposal standards”, radioactive waste blow the allowable clearance level radioactive waste concentration can be self-disposed. However, clearance level radioactive waste is not only controlled by “Nuclear Safety Act” but also controlled by other laws. It is described that the standards Article 8 (Restrictions on actions) 3. “Radioactive waste cannot be self-disposed in a manner prohibited by other laws applicable to non-radioactive materials contained in radioactive waste”.

In KAERI, about 11,000 drums (200 L) of uranium-contaminated soil were generated from decommissioning of uranium conversion plant. To lower the radioactive waste level and de-contaminate, we washed out the soil with 1 M sulfuric acid and satisfied the concentration of clearance level radioactive waste standards.

Therefore, the Nuclear Safety Act was satisfied for clearance level radioactive waste, and any violations of other laws were confirmed.

We have reviewed the laws and regulations related to the disposal of radioactive waste, specifically with regards to landfilling. The primary law that applies is the Waste Management Act. We have confirmed whether the waste is considered a designated waste according to the Enforcement Decree of the Waste Management Act, whether it is a toxic substance according to the notification of toxic substances designation, and whether it meets the standards for soil contamination concerns under the Enforcement Rules of the Soil Environment Conservation Act.

We used sulfuric acid to decontaminate the uranium from the waste, and we removed any residual sulfuric acid. As a result, we have determined that it does not fall under the aforementioned laws and regulations, and it has been approved for self-disposal in March 2024.

Keywords: Clearance, Radioactive waste, Self-disposal, Contaminated soil, Nuclear safety act

A Study on the Radioactive Concrete Waste Reduction and Recycling Technology for the Decommissioning of Nuclear Power Plants

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According to the IAEA in May 2023, the total number of nuclear power plants 436 units are in operation around the world. Of these, nuclear power plants 209 units have been permanently shut down in 22 countries around the world. In Korea, Gori and Wolsong No. 1 have been permanently shut down. In the future, it is expected that old nuclear power plants that have exceeded their design life will be permanently shut down and decommissioned. For this reason, technologies and regulations for the safe decommissioning of nuclear power plants should be prepared. The largest area of nuclear decommissioning technology is expected to be related to disposing of radioactive waste. The radioactive waste generated during the decommissioning of nuclear power plants is expected to amount to tens of millions of tonnes, including concrete, soil and metals and will require huge disposal costs. In particular, concrete waste accounts for more than 70% of radioactive waste and reducing the volume is essential to reduce disposal costs. Concrete waste reduction treatment involves physically and chemically separating the radioactive and non-radioactive contaminated portions to reduce volume and recycle, and many research is underway to develop technologies for this purpose. This study analyzed domestic and international research and practical applications of treatment technologies such as volume reduction, recycling, mechanical and chemical decontamination, and solidification of concrete waste to identify trends in technology development and derive improvement directions.

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Keywords: Concrete waste, Volume reduction, Heating and grinding, Nuclear power plants, Decommissioning technology

Evaluation of Applicability of Plasma Torch Melting for Perlite Heat Insulation Material

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KHNP CRI has developed and is operating the 100 kW class plasma torch melting facility to evaluate the applicability of the plasma torch melting. In this study, demonstration test was performed using the perlite heat insulation material as a simulant waste. And, to treat the perlite heat insulation material for plasma torch melting, additive for melt was developed using electric box furnace and slag prediction model. For the demonstration test, the temperature inside the plasma melter is approximately 1,600°C. The simulant was put into the melter by the vibrating feeding device with the throughput of 6 kg/hour. During the test, the power of plasma torch is 80–98 kW on the transferred mode.

Keywords: 100 kW class plasma torch melting, Perlite heat insulation material, Demonstration test, Applicability

Review on PE-HIC Generated From Contaminated Water Treatment Facility

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After the Fukushima accident, significant radioactive wastes have been generated and stored in the outside. Among of the radioactive wastes, the contaminated water is seawater and underwater containing not only salt component but also fission products. The contaminated water has been treated by Cs/Sr removal facilities (such as KURION, SARRY and SARRY II). In addition, ALPS system is operated to meet the regulatory standards for discharge. ALPS system is a facility of 62 radionuclide containing Cs and Sr except for Tritium and Carbon-14. Secondary wastes generated from these contaminated water treatment facilities is put into the PE-HIC (Polyethylene High Integrity Container) and stored in temporary storage facilities on the Fukushima Daiichi NPP site. Japan operates four culverts on the site. Among of the four culverts, PE-HIC with overpack container is stacked in three layers and stored in unit 2 and 3. There are 4,143 of the PE-HIC in temporary storage as of March 2023. TEPCO conducted an integrity assessment such as drop test of PE-HIC for handling in culverts.

Keywords: PE-HIC, Fukushima daiichi NPP, Culvert, Drop test

Case Study: A Bayesian Approach to Updating Scaling Factor for Radwaste

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To properly dispose of radwaste, it is necessary to assess its radioactive inventory. In Korea, the regulator allows for using an indirect method known as the scaling factor (SF) method for radwaste from nuclear power plants. An SF represents the concentration ratio of a specific difficult-to-measure (DTM) nuclide to its corresponding gamma-emitting nuclide. For thirteen DTM nuclides, including gross-alpha, their SFs are calculated as the arithmetic mean of measured SF values for a selective number of representative radwaste packages collected from each reactor group. A reactor group comprises reactors with the same reactor design such as CANDU 6. This method assumes that waste generated from the same reactor group possesses a similar radioactive inventory across all reactors within the group due to a similar waste generation process.

For each waste package for disposal, the concentration of each DTM nuclide can be indirectly determined by multiplying its SF by the concentration of the corresponding gamma-emitting nuclides measured for the package. The regulator mandates periodic validation of the derived SFs for each newly produced radwaste in each NPP group every two years. During this validation, the SFs, determined for packages selected from the newly produced radwaste in each reactor group, are determined whether they are within $\pm$ a factor of 10 of the derived SF for each DTM nuclide. So, if all the SF values are within the range, then the derived SF can be used for the newly produced radwaste; otherwise, a new SF should be established for the DTM nuclide in the reactor group in question.

This study presents the CERN (European Organization for Nuclear Research) case, where the SFs are updated by using the Bayesian approach, which can cope effectively with a more realistic situation that the radioactive inventory in a reactor group can change over time. The Bayesian approach can predict changes in the distribution of the scaling factors, as well as in the parameters defining the distribution over time, by combining the prior information (the distribution of SF before the production of new radwaste) and the data (the radiochemical analysis results for the newly produced radwaste) to derive posterior information on the distribution of SF.

Keywords: Scaling factor, Radwaste, Bayesian approach, Radioactive inventory

Study on the Cooling Hose of Vitrification Facility

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The Purpose of this study is to confirm the equipment integrity of the CCIM cooling system of Han-ul vitrification facility. The hose installed in the CCIM is made by NBR and is connected to the SPC plate, injector, camera, etc. These hoses supply coolant water and oxygen. Hose exposed to high-temperature cooling water in a vitrification facility for a long period of time may crack due to hardening, and the integrity of the hose is reduced. Damage to the hose occurs when the fitting and hose connection bend at a nearby point. To solve this problem, the hose material was changed in this study. By selecting new materials, the cooling hose integrity of the cooling system was confirmed.

Keywords: Vitrification, Cooling system, Hose

Development of Copper Tellurite Glass Composition to Immobilize Radioactive Iodine

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Radioactive iodine-129 can easily diffuse into the environment and possesses a half-life of 1.57×10^7 years. Therefore, durable and stable waste forms are required for the treatment I-129. In this study, $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glasses were fabricated by incorporating CuI into a copper tellurite glass ($0.3\text{Cu}_2\text{O} \cdot 0.7\text{TeO}_2$) matrix at different concentrations ranging from 0 to 30 mol%. The X-ray diffraction analysis confirmed the amorphous phase of all prepared samples, and X-ray fluorescence analysis verified iodine volatilization during the melting process. With the addition of 30 mol% CuI, the maximum iodine loading was approximately 13.79wt%. Furthermore, all samples satisfied the US regulation of 2 g/m² for the normalized release of all elements, as confirmed through the product consistency test. The proposed glass waste forms have cost advantages over previously developed silver-based glasses, making them a promising avenue for further research.

Keywords: Radioactive iodine, Copper iodide, Vitrification, Copper tellurite glass

A Case Study on the Off-Gas Treatment System in Radioactive Waste Treatment Facilities

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The Republic of Korea, various thermal, mechanical, and chemical treatment technologies are being researched and developed to treatment of radioactive waste generated from the dismantlement of Kori Unit 1 and Wolsong Unit 1. Among these, thermal treatment technology methods include incineration, pyrolysis, and melting. These technologies are effective in reducing waste volume, but there is a problem that radioactive materials (Cs-137, I-131, Sr-90, etc.) may be exposed to the atmosphere through exhaust gases.

Also in Korea, melting technology (plasma torch, induction heating, etc.) is being developed to dispose of radioactive waste, and along with melting technology, an off-gas treatment system is being developed that can solve the problems mentioned above and ensure acceptance by residents.

The off-gas treatment system is a device to minimize the air emission of gaseous substances, and the general industry uses equipment specified by the CLEAN AIR CONSERVATION ACT. In the field of radioactive waste treatment, research and development on facilities that minimize the emission of radioactive materials is needed along with the CLEAN AIR CONSERVATION ACT. For this reason, there are many proven off-gas treatment technologies in general industry, but not much information related to the radioactive waste treatment industry.

This paper contains a review of research and development cases of off-gas treatment systems applied to general industry and overseas radioactive waste treatment facilities in order to review factors related to off-gas treatment technology to be applied to domestic radioactive waste treatment technology.

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Keywords: Plasma, Melting, Radioactive waste, Waste treatment, Commercialize, Demonstration

A Case Study on Commercialization of Radioactive Waste Thermal Treatment Technology

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The Republic of Korea is preparing to dismantle the Kori Unit 1 and Wolsong Unit 1. The amount of radioactive waste generated when dismantling nuclear power plants is expected to be around 80,000 drums, and the goal is to reduce it to 14,500 drums to secure disposal costs and disposal space. As a result, various waste treatment technologies for various wastes (metal, liquid, mixing, activated carbon waste, etc.) are being researched and developed.

Waste treatment technologies include thermal, mechanical, and chemical methods, and the applicable technologies differ depending on the type of waste. Representative technologies for thermal treatment include plasma, pyrolysis, and incineration. Among these, plasma is a technology that can treat waste regardless of its type. There are cases overseas where it has been applied to radioactive waste treatment as a commercialized facility.

For example, the Plasma Melting Facility (PMF) located in Kozloduy, Bulgaria, conducted operational tests for each device of the melting system, and conducted safety operation tests for normal and abnormal operation for 72 hours. In addition, A melting test was conducted on simulated waste and commercialization test was completed. Also in Korea, various research and development are being conducted to commercialize plasma torch melting, and it is expected that elements such as testing each component of the system and safety tests for normal and abnormal operation will be required for technology commercialization.

This paper studied overseas commercial cases for the commercialization of radioactive waste treatment plasma torch melting, and reviewed the factors for commercialization of domestically developed equipment in the field of radioactive waste treatment.

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Keywords: Plasma, Melting, Radioactive waste, Waste treatment, Commercialize, Demonstration

A Conceptual Design for Treatment of Large Concrete Drums in Kori Unit 1

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The Kori unit 1 nuclear power plant (NPP), which is the very first one in Korea, was permanently shut down in last 19th June 2017. Since the Kori unit 1 is the first NPP in Korea, large number of radioactive wastes were generated, compared to current status. The radioactive waste management system was not clearly fixed at the early stage of operation. The boron concentrates, which is treated using concentrate waste drying system (CWDS), is solidified using cement at the early stage of NPP operation. Since radioactive nuclide inventory in boron concentrates is relatively high compared to current status, some of the cement solidified boron concentrates were re-packaged in large concrete drums. Since radioactive nuclide inventory, mechanical/chemical properties of the solidified materials, etc. are not specified, the large concrete drums, containing cement solidified boron concentrates, do not satisfy the current waste acceptance criteria (WAC) of KORAD, as it is.

In this study, segmentation of outer large concrete drum, separation of inner drum, re-packaging of inner radioactive waste will be discussed.

Keywords: Large concrete drum, HIC, Packaging, Radioactive waste

KRID's Physical/Radiological Characterization Plan for Intermediate-Level Decommissioning Waste

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Kori Unit 1, Korea's first pressurized water reactor nuclear power plant and Wolsung Unit 1, the first pressurized heavy water reactor nuclear power plant, were decided to be permanently shut down in 2017 and 2019, respectively. KRID was established in 2020 to restore the environment through stable decommissioning of used nuclear power plant. KRID is an industrial development and research organization that leads domestic nuclear decommissioning technology independence and seeks to serve as a hub for nuclear decommissioning. However, there are only four countries in the world that have successfully decommissioned nuclear power plant: the United States, Germany, Japan, Switzerland. In addition, information such as nuclear power plant decommissioning technology and nuclide distribution for each component was not disclosed. Accordingly, KRID seeks to establish an analysis infrastructure dedicated to intermediate-level decommissioning waste that can analyze the distribution and amount of nuclides in various parts, especially intermediate decommissioning waste, build a database, and perform physical/structural characteristic analysis of materials. The dedicated infrastructure for intermediate-level decommissioning waste scheduled to be built at KRID will consist of a concrete hot-cell and a laboratory for nuclide analysis. The concrete hot-cell consist of a total 7cells, and within the cell, specimen processing for characterization, physical characterization, and preprocessing for nuclide analysis will be performed. The laboratory for nuclide analysis seeks to measure and analyze alpha, beta, and gamma on intermediate-level waste that has been pretreated in a hot-cell. Lastly, KRID aim to secure data on the physical characteristics of decommissioning waste and identify 95% of nuclides to satisfy suitability for acquisition and disposal. To this end, 47 types of characterization, processing, and pretreatment equipment will be built, and the facility is scheduled to be completed within 2026. Based on this foundation, KRID aims to lead the safe and efficient decommissioning of domestic nuclear power plants and further contribute the the development of the global nuclear power plant decommissioning industry.

Keywords: Intermediate-level radioactive material, Decommissioning waste, Hot cell, KRID

A Conceptual Design for Pre-Treatment Equipment for Large Concrete Drums in Kori Unit 1

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The large concrete drums generated and stored in Kori unit 1 nuclear power plant (NPP) will be properly treated to satisfy the waste acceptance criteria (WAC) of KORAD. Since the inner radioactive drum will be re-packaged and/or re-solidified, the segmentation, chipping, and additional processes are prerequisite. The processed will be implemented in a radiation-controlled area. During the suggested, indicated above, processes, the radiation exposure to worker and generation of secondary wastes are expected. The shielding house design offers the effective separation of operation space and prevents the unnecessary radiation exposure and spread-out of secondary waste. The segmentation tools are also important aspect when consider the waste treatment. The pilot test based variable consideration for segmentation of concrete drums allows the practical design of the tools.

In this study, the conceptual design for pre-treatment equipment for large concrete drums will be discussed. The equipment includes shielding housing, segmentation tools, chipping tools, local ventilation system, etc.

Keywords: Large concrete drum, Pre-treatment, HVAC, Secondary waste

A Review on Treatment of Non-Fuel Materials in the SFP of PWR

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PWRs (Pressurized Water Reactor) being operated in Korea are classified into WH type, OPR type, and APR1400 type. Kori 1, 2, 3, 4, Hanbit 1, 2 and Hanul 1, 2 correspond to WH type nuclear power plant (NPP). Except for CANDU plants, the remaining plants are OPR1000 and APR1400 type NPP. The OPR1000 and APR1400 have similar operating methods, so non-fuel materials are managed similarly. Because WH type NPP's operating method is different from OPR1000 and APR1400, non-fuel materials are differently managed in viewpoint of Spent Fuel Pool (SFP) operation. Typically, Burnable poison rods, CEAs (Control Element Assembly) and thimble plugs are non-fuel materials generated only in WH type NPP and most of them are stored in SFP with being inserted into SF. Although spent CEAs are generated in domestic PWR including OPR1000, APR1400 and WH type NPP, CEAs generated in WH type NPP are stored with being inserted in the spent fuel. Whereas OPR1000 and APR1400 NPP's CEAs are stored in a separately managed basket. KHNP, meanwhile, recently launched a general design service for dry storage system and facility. The dry storage system is a MAGNASTOR cask that NAC has been approved by NRC. According to approval, this cask can store non-fuel materials such as ICI and CEA inserted into the fuel together. Since KHNP plans to design a dry storage system using the design of this cask in the future, it is judged that WH type fuel, which already includes CEAs and poison rods, can be stored in these casks. However, in case of OPR100 and APR1400, it is necessary to deduct a method to contain CEAs and ICIs into the fuels. When considering the treatment method of non-fuel materials in the future, these measures should also be considered.

Keywords: Non-fuel material, ICI, CEA, Treatment, Saturation

Experimental Study of the Adsorption Behavior of Iodine and Technetium on Dry Active Waste

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Predicting the long-term behavior of radioactive nuclides after closure of the waste disposal facility is critical for establishing the safety assessment system. Considering the various types of waste that are expected to be stored in the disposal facility, accurately categorizing them and deriving the behavior of nuclides is important because it directly informs source term modeling. Especially, the dry active waste, which constitutes a substantial portion of wastes in disposal facility, it is important to obtain data on the leaching behavior of nuclides for its own medium. However, several researches have been focused on natural and engineered barrier systems, only few studies for dry active waste. This lack of characteristic data of the dry active waste may lead to inadequate representation of actual waste in safety assessments. Thus, it is necessary to derive the waste characteristics according to the types of waste.

Tc and I are considered key radionuclides in safety assessment due to their long half-life and high dose contribution. In addition, research on bentonite and soil indicates that Tc and I are highly mobile nuclides because of their very low retardation factor. This mobility can be affected by not only the chemical conditions of the disposal facility but also the physical characteristics of waste, such as hydraulic conductivity, diffusion coefficient and adsorption coefficient (K_d). K_d is one of the key factors in identifying the nuclide leaching since the retardation effect can be occurred by adsorption when waste is saturated with groundwater and follows diffusion or advection behavior. The adsorption behavior can be changed according to the various conditions such as the pH and temperature, thus it is important to experiment under the actual disposal waste and environment. Therefore, the adsorption experiments of Tc and I were performed with dry active waste which take up the large portion of disposal facility waste.

In this study, batch adsorption test was conducted to investigate the adsorption behavior of Tc and I. Rhenium (Re) was applied instead of Tc due to its similar chemical behavior, and non-radioactive NH_4ReO_4 and NaI were used as surrogates for Re and I, respectively. Dry active waste was simulated using fresh materials which can be expected in actual used in nuclear power plants, categorized into textile, plastic, and rubber wastes. As the result of this study, the experimental data can be utilized in dry active waste modeling for safety assessment.

Acknowledgements

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Keywords: Dry active waste, Low-Intermediate level waste, Adsorption coefficient (K_d), Batch adsorption test

Adsorption Behavior of Iodine and Technetium on Calcium-Silicate-Hydrated (C-S-H) Phases

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Cement is one of the widely used solidification material for radioactive waste, and extensive research has been conducted on adsorption or leaching within cementitious media. Evaluating the adsorption of various nuclides on cement phases is important for identifying the main radionuclide immobilization mechanisms and providing key information of leaching behavior. Calcium Silicate Hydrate (C-S-H) phases, generated during cement hydration, take up a large portion of the cement solidified waste and known as immobilizing radionuclides effectively. For example, Cs can be adsorbed through electrostatic interaction between the surface and ionic charges based on the Ca/Si ratio. Similarly, Sr can be immobilized through ion exchange with calcium ions in the ion-exchangeable interlayer of C-S-H. In case of Co, can undergo adsorption by both electrostatic interaction and ion exchange into C-S-H, and known as existing in oxide form.

Research on such adsorption mechanisms primarily focuses on cationic radionuclides, given that the majority of radioactive nuclides are cations. However, it is necessary to consider anion radionuclides such as Tc and I, which are of interest due to their high mobility and long half-life, significantly impacting the safety assessments. This suggests that the study of adsorption I and Tc can cause the results of the safety assessments and affects to waste acceptance criteria. Therefore, in this study, the adsorption mechanism of C-S-H, the main component of cement solidified waste, was analyzed by batch adsorption experiments.

Batch adsorption tests were conducted to investigate the adsorption behavior of Tc on C-S-H gel. Sodium Iodide (NaI) was used as it generally exists in the form of I⁻ under alkaline conditions. Non-radioactive Rhenium (Re) served as a surrogate of Tc due to its similar chemical behavior and ionic radii. NH_4ReO_4 was used as surrogate to simulate TcO_4^- , which can be expected form under disposal condition. C-S-H was synthesized by mixing CaO and nano-SiO₂ to achieve the target Ca/Si molar ratios of 0.81, 1.0 and 1.5, to analyze electrostatic adsorption effect on Tc and I. Adsorption mechanism was discussed by radionuclide concentration and time intervals. As the result of this study, data on the adsorption mechanisms of radionuclides by C-S-H can be obtained which is the main component of cement solidified waste, and serve not only as fundamental data for understanding cement solidification waste research but also as a basis for further studies.

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Keywords: Calcium-silicate-hydrated (C-S-H), Cement solidified waste, Adsorption coefficient (Kd), Batch adsorption test

Status on Clearance of Decommissioning Waste at KRR-1 and -2

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The decommissioning of Korea Research Reactor Units 1 and 2 (KRR-1&2), the first research reactors in South Korea, began in 1997 and the decommissioning status is currently proceeding with phase 3. For KRR-1&2, the unrestricted release of the site is in progress through the decommissioning of remaining buildings, a final radiation survey of the site, decontamination, site restoration and so on. In addition, it is important that disposal of radioactive waste within the site must take place prior to the final radiation survey and site restoration. Accordingly, the disposal of radioactive waste has been accelerating from 2021. Especially, for various reasons such as economic effectiveness and disposal site capacity, many efforts are being made to reduce amount of radioactive waste by encouraging clearance only for wastes with minimal radiological risks at KRR-1&2. From 2021 to present, a total of 5 cases of clearance have been approved, and 1 additional case is currently under review by the regulatory agency. In this study, the clearance results will be described in the case of concrete, metal, asbestos at KRR-1&2.

According to HPGe analysis results, gamma-ray emitting nuclides such as Co-60, Cs-137, and Eu-152 were commonly detected in clearance wastes at KRR-1&2. Furthermore, it was confirmed that H-3 and C-14 were also detected in some wastes and gross-alpha, -beta were measured using a Low Background Alpha/Beta Counting System. The identified maximum concentrations were 3.08E-02, 2.20E-02, 3.15E-02, 4.82E+01, 2.80E-01 Bq/g for Co-60, Cs-137, Eu-154, H-3, and C-14, respectively. H-3 and C-14 were detected only in concrete waste, and the gross-alpha and -beta values in other types of waste were all below MDA. Although the concentration of major nuclide in concrete waste was observed to be relatively high, it was finally approved as it satisfied the sum of fractions (less than 1) in accordance with the Nuclear Safety Act. In present, approximately 101 tons of concrete waste are being reviewed, and attempts for clearance of waste will continue to be made for metal and solid waste.

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Keywords: Clearance, Decommissioning, KRR 1&2, Radioactive waste

Safety Analysis on the High-Temperature Catalytic Oxidation Process for the Treatment of Spent Scintillation Liquid

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The presence of organic components in spent scintillation liquid should be considered during all steps of radioactive waste processing for final disposal. Scintillation liquids often referred to as cocktails are generated from radiochemical analyses of radionuclides, which mainly consists of mixtures of liquid organic materials such as toluene and xylene. Typical features of these liquid organic materials are volatility, combustibility and toxicity. These are the reason why special attention must be paid to the management of liquid organic radioactive wastes. To select an appropriate waste management strategy and to design the treatment process of spent scintillation cocktails, authors investigated the specific radioactivity and moisture content of spent scintillation liquid generated at Wolsong nuclear power plants in the previous study. Among the various available technologies, the high-temperature catalytic oxidation technology was selected for the treatment of spent scintillation liquid in this work. Based on the above results, the safety analysis on the high-temperature catalytic oxidation process will be performed and the results will be discussed.

Keywords: Scintillation liquid, Scintillation cocktail, Organic radioactive wastes, Liquid scintillation counter, Liquid scintillator

~~Preliminary Study on an Uncertainty Evaluation of Chemical Pretreatment on Radioactive Waste~~

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Withdrawal

A Preliminary Study on Deriving Implications for the Clearance of Decommissioning Waste in Korea Through International Case Studies

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There has been an increase in facilities using Radioisotopes for various purposes such as medical, industrial, and research activities in Korea, leading to a continuous increase in the generation of Radioisotope (RI) wastes. However, the majority of RI waste is not being delivered for final disposal. The issue stems from the characteristics of the RI waste generation. As RI wastes are used for various purposes across different fields, its volume and timing of generation are highly irregular, and its characteristics are complex and diverse. Consequently, there has been a lack of specific guidelines for the delivery, acceptance criteria, disposal methods, and timing for such waste.

There is a growing need to present a revised classification and management plan that specifies the acceptance criteria for RI waste, to facilitate the smooth transfer of waste and its efficient management. This study has investigated and compared the domestic and international classification systems and IAEA recommendations to derive a management system for RI wastes. The radioactive classification system and management methods in France and Sweden were investigated and analyzed for applicability to domestic RI waste. The D-Value used in the classification of radioisotopes in IAEA RS-G-1.9 was examined to see if it could be applied to the classification of RI waste in Korea, and a management plan was derived.

The proposed improvement plan aims to maximize the amount of waste that can be recycled, self-disposed, or stored until decay to a manageable level for self-disposal, thus reducing the burden on disposal contractors. This approach is expected to increase flexibility in waste handling and allow disposal contractors to focus more efficiently on managing higher-risk waste categories.

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Keywords: Decommissioning waste, Clearance, Radioactive waste

Bench-Scale Testing for the Development of a Commercial-Scale Paddle Type Heating-Grinding Integrated Reactor

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This study aims to develop a dry separation system for volume reduction and radionuclide removal of concrete waste based on a paddle type heating-grinding integrated reactor, with the objective of reducing the quantity of radioactive waste by heating-grinding the radioactive concrete waste to separate cement paste, which contains radioactive contaminants, and aggregates that are relatively less contaminated and can be recycled. We previously designed a unit equipment based on the rotary mill type method to develop an integrated reactor capable of simultaneous heating-grinding of concrete waste. However, we found that connection with other equipment was necessary after tilting during insertion and extraction of concrete waste. Consequently, we concluded that it would be difficult to achieve complete control over radioactive dust and gas using the method applied in the initially designed equipment. Complete control over radioactive dust and gas is essential in this system that handle and process radioactive materials. Thus, the heating-grinding integrated reactor was redesigned based on a paddle type method, allowing for the entire process to be enclosed.

Prior to the development of a commercial-scale heating-grinding integrated reactor based on the paddle type method, we first fabricated a bench-scale equipment and conducted cold tests through operation. Subsequently, we validated the applicability of this technology by comparing its performance with that of bench-scale and pilot-scale rotary mill equipment previously developed. Additionally, we identified engineering improvements and operational know-how for the fabrication and process establishment of the commercial-scale heating-grinding integrated reactor through preliminary verification.

This equipment can effectively reduce radioactive waste through dry separation of cement paste, which is radioactive contaminant, from radioactive concrete waste. In addition, it has the advantage of minimizing worker exposure by completely control radioactive dust and gas. It is expected that the development and commercialization of a concrete waste heating-grinding dry separation system based on this technology will contribute to the advancement of recycled aggregate recovery technology for over 50 million ton/year of concrete waste generated in South Korea. Furthermore, it is anticipated to contribute to addressing the volume reduction and self-disposal issues of radioactive concrete waste, which are expected to sharply increase during the decommissioning of nuclear-related facilities.

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Keywords: Paddle type heating-grinding integrated reactor, Radioactive concrete waste, Dry separation, Volume reduction, Self-disposal

Thermodynamic Equilibrium Calculations of Treatment of Radioactive Wastewater Containing Uranium

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An acid washing process can be applied to clean uranium-contaminated soil, and radioactive wastewater with various ions including uranium is generated during this process. It is necessary to derive a method that can remove only uranium from the wastewater for minimizing radioactive waste generation. For this reason, thermodynamic equilibrium calculations of treatment of radioactive wastewater containing uranium were performed. The treatment concept was based on a reaction between uranium and hydrogen peroxide. Basic additives were utilized to promote the reaction in the calculations. As a results of the calculations, it was expected that uranium can be precipitated in an insoluble form in the wastewater at high efficiency. This efficiency can depend on the basic additives. It is thus necessary to select an appropriate basic additive considering the efficiency and waste generation. This result will be utilized to remove only uranium from the wastewater.

Keywords: Acid washing, Radioactive wastewater, Uranium, Uranium-contaminated soil, Thermodynamic equilibrium calculation

A Study on the Management Plan of Radioactive Waste Management System for Continued Operation of Nuclear Power Plant

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Continued operation of long-term operating nuclear power plants is underway worldwide. In order to continue operating, a safety evaluation must be carried out in various fields and reviewed and approved by the regulatory body. Therefore, this study will discuss considerations for the field of waste management. In order to effectively manage radioactive waste management systems for continued operation of nuclear power plants, policy and regulatory compliance, establishment of an integrated management system, risk assessment and management, monitoring and monitoring systems, etc. should be considered.

Radioactive waste management systems must comply with relevant policies and regulations, including the International Nuclear Safety Regulatory Organization and national statutes and regulations. Radioactive waste management requires an integrated and systematic approach. This requires integrating various management systems, efficiently managing data, and establishing a system that can be used for decision-making. Radioactive waste poses a potential risk, which requires risk assessment and management. This allows you to take steps to identify and minimize risks.

A system must be established to continuously monitor and evaluate the condition of radioactive waste. For this purpose, sensor networks, sensing equipment, and surveillance systems can be utilized. The research and development of radioactive waste treatment technology must be continuously promoted. This allows us to develop technologies that can safely treat and finally dispose of radioactive waste.

Workforce involved in the management of radioactive waste must receive professional training and training, which enables safe operation and management.

Through these measures, a radioactive waste management system for continued operation of nuclear power plants can be effectively managed. Therefore, this study intends to present the detailed review results of the management plans for the above four fields.

Keywords: Continued operation, Radioactive waste, Management, System, Regulation

Development of Prototype of Handling Device for Large Concrete Drums in Kori Unit 1

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In the Kori unit 1 of radioactive waste storage facility, large concrete drums are stored for a long time including concentrated waste, spent resin and spent filter. The large concrete drum was generated before waste acceptance criteria (WAC) was established. To treat and disposal the containers with radioactive wastes, it is necessary to develop technologies for a characteristic evaluation of large concrete drum with radioactive wastes and handling. There are two types of large concrete drum in the Kori unit of radioactive waste storage facility. Square-type drum is 1,460 mm (width), 1,460 mm (length) and 1,180 (height) and weighs less than 6 tons. There are four 200 L-drums in the square-type drum. For round-type drum is 1,060 mm (diameter), 1,370 mm (height) and weighs less than 3 tons. The round-type drum is loaded with one 200 L-drum. Accordingly, in order to handle heavy and large concrete drums, a special device should be developed considering the space of the storage facility, the design of the drums, worker's radiation exposure and so on. In this study, the prototype of handling device for large concrete drums was investigated. And the demonstration test is being conducted with mock-up.

Keywords: Handling device, Large concrete drum, Prototype, Demonstration test

Development of on Evaluation Platform to Advance Radioactive Waste Solidification Analysis Technology

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A variety of radioactive waste is generated from domestically operated nuclear power plants, research institutes, industrial sites, etc. Radioactive waste is ultimately collected and disposed of in solid form. Radioactive waste whose safety is not guaranteed is processed in solid form through solidification and then disposed of. As a lot of time and money are consumed in the process of analyzing and evaluating whether the solidified body is safe, there is an urgent need to develop technology that can quickly generate and evaluate the properties of the solidified body.

To date, there has been no case of disposing of solidified waste at a domestic disposal site, and this is because reliability regarding the evaluation of solidified material characteristics has not been secured. Research is being conducted to introduce radioactive waste management technology using artificial intelligence and big data technology, but this is only for the purpose of using X-rays to determine the contents of radioactive waste and classify and manage them by period and item.

It is necessary to promote the development of standard evaluation technology for solidified materials that leads to non-destructive artificial intelligence analysis from solidification medium evaluation big data secured through the development of a solidified material characteristic evaluation platform and establish a solidification processing technology development structure to secure world-leading technology in the field of solidified material evaluation in the future.

Keywords: Solidified material, Disposing, Material characteristics, Artificial intelligence technology, Non-destructive

A Study on the Repackaging of Solidified Concentrated Liquid Radioactive Waste Into PC-HICs

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The concentrated liquid waste generated from PWR's consist of borate compounds mainly, and the concentrates generated in the 1980s were solidified under the Radioactive Waste Solidification Process Control Program.

However, the solidified concentrated liquid radioactive waste (solidified concentrates) produced in this way does not meet the requirements for disposal and acquisition of the waste disposal site and is being stored in NPPs. Various measures are under consideration to deal with these, and as one of methods, repackaging them into the polymer concrete high integrity container (PC-HIC) is considered.

In this study, we attempted to propose a method to effectively repack solidified concentrates in a 200 L drum into PC-HIC. Since 860 L PC-HIC is currently licensed as a disposal container for dried liquid waste powder, we considered a repackaging plan based on that container.

Since the inner diameter of the 200 L drum is approximately 567 mm and the inlet diameter of the PC-HIC is about 510 mm, the solidified concentrates from the 200 L drum cannot be loading into the PC-HIC as a whole. Therefore, we considered cutting the solidified concentrates of the 200 L drum into 90°, 60°, 45°, 30°, and crushing it and loading it into the PC-HIC, respectively.

We evaluated the optimal solution by comprehensively considering the cost of purchasing PC-HIC containers, container filling rate, waste disposal cost, work performance cost, and secondary waste.

In the case of not crushing the solidified concentrates, the method of cutting and loading them at 90° degrees was evaluated most effectively, and in the case of crushing, it was judged that further review was necessary in consideration of the cost of purchasing and installing the jaw crusher equipment.

Keywords: Liquid waste concentrates, Solidified concentrates repackaging

Effect of Salt in Radioactive Wastewater on the Stability of Cement Waste Form

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Wastewater containing radioactive nuclides and salt is generated from a nuclear facility. Hydraulic cement can be an attractive binder for the immobilization of low and intermediate level radioactive wastes. The wastewater can be stabilized using a cement solidification process. It is important to evaluate the stability of cement waste form for final disposal. Salt in the wastewater affects the stability of cement waste form. For this reason, effects of salt in a radioactive wastewater surrogate on the stability of a cement waste form were evaluated in this study. Na_2SO_4 , NaNO_3 and NaCl were used as a salt in the surrogate considering compositions of radioactive wastewater from the nuclear facility in KAERI, and Ordinary Portland Cement (OPC) was used as a cementation material. For the cement solidification, the contents of waste and salt were changed from 20 to 30wt% and from 0 to 12wt%, respectively. The stability of cement waste forms was tested after curing for 7 days, and it was evaluated using compressive strength and leaching tests. Through the cement solidification tests, a range of the salt content was derived to fabricate a stable cement waste form. This result will be utilized to determine a fabrication condition for a stable cement waste form of radioactive wastewater.

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Keywords: Cement waste form, Wastewater, Salt, Stability (compressive strength and leaching test)

Introduction of the Developed Paraffin Solidified Concentrate Waste Treatment Method and System

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In the past, the concentrated waste (CW) generated from Kori nuclear power plant (NPP) used to be solidified with paraffin, but the paraffin solidified CW (PSCW) did not satisfy leachability index of domestic waste acceptance criteria. These wastes are temporarily stored within the NPP, occupying the storage space and burdening the facility. Therefore, PSCW should be re-treated to meet solidification criteria for disposal. This paper introduces the treatment method and system developed for this purpose.

The developed treatment method aims to separate paraffin and CW to reduce the volume of radioactive waste for disposal. Considering the properties of paraffin and CW, they could be separated by heating PSCW in water. Considering melting point and density difference, paraffin would melt and float above the water, while CW would dissolve into water or settle down. The separated paraffin is expected to have low or no radioactivity, making it suitable for disposal as clearance waste. The separated CW could be dried via CW drying system installed at the NPP, then be solidified by packaging into high integrity container or mixing into a stable matrix, such as cement.

Paraffin withdrawal rate of above treatment method is experimentally determined. Lab-scale PSCW is manufactured by mixing melted paraffin and boron powder at weight ratio of 25:75. Then, the manufactured waste is placed into a beaker containing tap water of a mass 3 times that of added waste. Water is heated and stirred at 95°C for 2 hours to melt and separate paraffin, then cooled to solidify the paraffin for withdrawal. The withdrawn mass of paraffin increased by 4.05%, indicating a withdrawal rate of 95.95%. This rate could be increased up to 99% by adding NaOH to increase pH of water around 10, or re-melting separated paraffin into water 2~3 times.

A PSCW treatment system is developed incorporating above treatment method. The developed system consists of a main heating tank (MHT) for paraffin separation, an auxiliary heating tank to heat-up make up water before adding it into MHT to minimize temperature decrease, and a storage tank for store ready-conditioned make-up water. PSCW and water are added to MHT to be heated and stirred. After melting, the side valve of MHT installed 1 mm above the estimated paraffin-water contact is opened to withdraw melted paraffin. Pilot test has successfully been conducted, and improvements to optimize the system has been determined, which are planned to be further researched in future.

Acknowledgements

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Keywords: Concentrated waste, Paraffin solidification, Waste treatment, Volume reduction

Experimental Study of Dissolution of Glass Fiber Waste in the Domestic Repository and Its Estimated Colloidal Impact

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Colloid migration has been issued to enhance radionuclide release from the radwaste repository, as radionuclides may sorb onto mobile colloids, and drift along with the colloidal motion instead of being sorbed onto immobile surfaces. Recently, glass fiber wastes (GFWs) has found to be crushed into small particles, thus raising the concerns of colloidal formation from GFWs. This study assesses the estimated colloidal impact of GFWs disposal within the domestic repository through literature review and experimentation.

According to SKB's reports, under the hyperalkaline environment of the cementitious repository, colloid formation may be promoted by various mechanisms, including the decomposition of organic materials, dissolution of inorganic materials, and corrosion of metals. Another experimental evidence further supports the dissolution of GFWs under hyperalkaline conditions, thus raising the concerns about colloid formation in the domestic repository, where pH 12.4 is approximately anticipated.

Experiment was conducted to investigate the dissolution rate of GFWs under domestic repository condition. The experimental solution was firstly conditioned to simulate the predominant ions (Na, Ca, K) in the domestic repository groundwater, then was conditioned with $\text{Ca}(\text{OH})_2$ and KOH to simulate hyperalkaline condition of cementitious repository. The pH was conservatively set between 12.5~13.0 to maintain pH above 12.4 during heating. Average of 60 mg/L of GFWs and 150 mL of the experimental solution were added to PTFE containers, then the samples were separately allocated into heating baths, each set at 95°C and 25°C.

After 1 month, GFWs were filtered and recollected to measure mass change, then the measurement was substituted into the Arrhenius Equation to calculate the acceleration factor (AF) of the experiment. The AF was calculated to be about 226, meaning that the 1-month experiment's result represent 18.83 years at 25°C. By substituting the AF back into the Arrhenius Equation, complete dissolution of GFWs was calculated to take 33.45 years, indicating GFWs would completely dissolve within the institutional control period, thus colloidal issue is unlikely.

However, the measured Si concentration in the experimental solution was low compared to dissolved mass of GFWs. This discrepancy may be attribute to precipitation of Si through the C-S-H reaction, which forms gel-like precipitates and promote sedimentation, particularly in high Ca^{2+} concentration. Considering gel-like precipitation formed at the bottom of the PTFE container after the experiment, C-S-H reaction could have occurred, explaining low Si concentration. In future, further study is planned to fully understand the detailed colloidal impact of GFWs disposal in the domestic repositories.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. RS-2303-00234190).

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

A Study on Material Integrity Through Biodegradability of PE-HIC

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High Integrity Containers (HIC) are containers capable of disposing of radioactive waste (such as sludges, concentrated liquid wastes powders, etc.) without solidification treatment, using general steel drums. To dispose of PE-HIC containers, it is necessary to demonstrate the integrity of PE material within disposal facility environments according to disposal facility acceptance criteria.

This study aims to assess the material integrity through the biodegradability of polyethylene (PE) material. According to domestic and international research, PE is highly non-biodegradable and inert, making it extremely difficult to decompose in the environment even after hundreds of years of burial. Previous PE biodegradation studies have investigated changes in physicochemical properties and structural degradation using microorganisms, UV, etc. However, it is widely acknowledged that most of these studies are insufficient to fully demonstrate biodegradability.

Keywords: Polyethylene, High integrity containers, Biodegradation

A Study on Overseas Disposal Cases of PE-HIC

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High Integrity Containers (HIC) are vessels designed for the disposal of radioactive waste (including sludges, concentrated liquid wastes, powders, etc.) without the need for solidification treatment typically mandated for conventional steel drum packaging. Currently, domestic nuclear power plants use HIC made of polymer concrete (PC) material. However, there is a situation where radioactive waste stored in High Density Polyethylene (HDPE) material HIC, generated in the past, also needs to be disposed of. In domestic nuclear power plants, PE-HIC is generated and stored for the storage of intermediate and low-level radioactive spent resins. There have been no disposal cases of HIC domestically yet, and the approval process for PC material HIC is currently underway, with research planned to commence for the disposal of HIC made of Polyethylene (PE) material in domestic disposal facilities.

The US NRC has evaluated PE-HIC as unsuitable for standalone use according to design criteria due to aspects such as creep and buckling, transition from ductile to brittle behavior, pre-irradiation effects, and degradation of mechanical properties over time. Consequently, PE-HIC's standalone usage has not been approved, and it is currently being disposed of at the Barnwell disposal site by over-packing with containers made of concrete material.

Keywords: Polyethylene, High integrity containers, Biodegradation

A Study on Technical Standards for Packaging Spent Filters by UK NIREX

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UK NIREX, responsible for managing low to intermediate level radioactive waste, has undertaken tasks such as safety evaluations of disposal facilities and establishing standards for the handling and disposal of radioactive waste. Currently, it is incorporated into the NDA and continues its operations. Technical Standard WPS/905 (Guidance Note on the Packaging of Filters, June 2006) proposes packaging approaches for waste filters based on acceptance criteria for phased geological repository concept. This standard applies to all types of filters used in gas and liquid purification systems. Depending on the characteristics of the waste filters (shape, material, radioactive properties, etc.), a classification system has been established, presenting four packaging approaches: Compaction, In-situ Conditioning, Grout Enclosing, and Destruction. Compaction involves compressing the filters to achieve compaction fixation, which is most efficient in terms of volume reduction, minimizing porosity, and waste immobilization. In-situ Conditioning involves using suitable polymer conditioning agents to penetrate the filters tightly, immobilizing radioactive particles, and then encapsulating them. Grout Enclosing entails placing the filters directly into waste containers and sealing them with cement grout or polymer, without penetration. Finally, Destruction involves partially or completely dismantling the filters, classifying them, and immobilizing the materials appropriately.

Keywords: Filters, NIREX, PGRC, GWPS

A Study on Effect of Hf Addition in DG-2 Glass

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Korea Hydro & Nuclear Power (KHNP) had commercialized the world first vitrification treatment technology for intermediate and low-level waste in 2009. The technology dramatically reduces the volume of waste, enhances the disposal stability, etc. The target waste of the technology is dry active waste (DAW), spent resin, and their mixture. The borosilicate glass, DG-2, was suggested for the vitrification of DAW. The DG-2 exhibits good processability: viscosity and electrical conductivity. The Hafnium (Hf) is a tetravalent transition metal. It is expected that the Hf in borosilicate glass will strengthen the binding between inorganic components and increase the stability.

In this study, effects of Hf addition in DG-2 glass is systematically investigated. The structural evolution and characteristics will be reported.

Keywords: Vitrification, DG-2, Hf, Composition

Solidification of Soil Containing Boric Acid Using Portland Cement

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Contaminated soil with small particle size is a representative type of dispersible waste. Dispersible waste must be converted into a physically stable form in accordance with the waste acceptance criteria for final disposal. Cement is general material used for solidification and immobilization of radioactive waste. However, components in the contaminated soil can affect the properties of solidified form generated using cement. For this reason, the effect of boron contained in soil on the physical property of solidified form was investigated in this study.

In order to verify the effect of boron, simulated soil wastes containing different proportions of boric acid was prepared. In order to prepare the simulated soil wastes, boric acid with a similar particle size to soil was mixed with the soil using vibrating machine. The solidified specimens were prepared by mixing same ratio of water-cement-simulated soil waste. The concentration of boric acid in the simulated soil was different. The specimens are cured for 7 days at room temperature. During curing, the hardening process of specimens were visually observed. After curing, the physical properties of solidified form were analyzed by compressive strength test.

As a result, it was observed that hardening of specimen was delayed as the concentration of boric acid in the simulated soil waste increased. In addition, the compressive strength of solidified form decreased when the amount of boric acid in the simulated soil waste increased. These are caused by a decrease in pH as the amount of boric acid in the soil increases. It is also believed that the compressive strength of solidified form decreased because the hydration reaction of the solidified form was interrupted by boric acid. The results of this study can contribute to establishing the basic conditions for solidification of dispersible waste including boric acid.

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Keywords: Cement, Solidification, Soil, Boric acid

Structural Analysis of Copper Tellurite Glass for Radioactive Iodine Immobilization

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Radioactive iodine-129 has a half-life of 1.57×10^7 years and can easily diffuse into the environment. In this study, $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glasses were fabricated to immobilize radioactive I-129 generated from nuclear facilities. The XRD analysis confirmed the amorphous phase of all samples, indicating successful glass formation. The maximum loading of iodine in the glass samples was approximately 13.79wt% when 30 mol% CuI was added in the samples. X-ray photoelectron spectroscopy and Raman spectroscopy were used to understand the structure of the $\text{CuI} \cdot \text{Cu}_2\text{O} \cdot \text{TeO}_2$ glass waste forms. As the CuI content increased in the glass matrix, there was an increase of +2-valent Cu and O_{BO} . It was further confirmed that there were no significant changes in the structure of Te, which serves as the glass network former. Furthermore, all samples satisfied the US regulation of 2 g/m^2 for the normalized release of all elements, as confirmed through the product consistency test. The proposed glass waste forms have cost advantages over previously developed silver-based glasses, making them a promising avenue for further research.

Keywords: Structural analysis, Radioactive iodine, Copper tellurite glass

A Review of Geopolymer Properties According to Composition and Reaction Conditions

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Nuclear Power Plants (NPPs) in Korea store a large amount of radioactive waste, including boric acid and spent resin. Boric acid and spent resin are classified as intermediate and low level waste (ILLW). In order to dispose of boric acid powder and spent resin, it must be manufactured in a durable waste form. Recently, geopolymer waste forms have been developed to immobilize radioactive waste. In particular, the geopolymer could encapsulate boric acid or spent resin and have high physical strength. In this paper, the properties of the geopolymer according to the composition and reaction conditions were reviewed. This review paper could be the basis for boric acid powder and spent resin treatment in Korea.

Keywords: Geopolymer, Radioactive waste, Boric acid, Spent resin

A Study on Solidification of Radioactive Waste From Nuclear Power Plant

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The immobilization of radioactive waste is crucial for long-term environmental and human health protection. This study explores various methods such as glassification, polymerization, cementation, paraffin embedding, and geopolymerization for effectively containing and isolating radioactive wastes. Glassification transforms radioactive wastes into stable glass matrices, while polymerization utilizes polymer networks for enhanced mechanical strength and adaptability. Cementation forms solid matrices to trap radioactive wastes, and paraffin embedding aids in handling small-scale wastes in laboratory settings. Geopolymerization introduces environmentally friendly binders for efficient immobilization. These methods collectively contribute to comprehensive waste management strategies, ensuring the safe treatment and isolation of radioactive waste.

Keywords: Radioactive waste, Solidification, Waste treatment strategies

A Review of Geopolymer Waste Form for Radioactive Waste Immobilization

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Safe disposal of radioactive waste is a critical challenge in nuclear waste management. Recent research highlights geopolymer waste forms as promising alternatives for effectively immobilizing radioactive materials. Geopolymer waste forms encapsulate radioactive wastes efficiently through stable matrices made of geopolymers, minimizing environmental contamination. Geopolymer waste forms exhibit excellent chemical resistance and low permeability, ensuring effective immobilization of radioactive wastes. With potential advantages in sustainability and cost-effectiveness, geopolymers offer promising prospects for future waste management strategies. This review examines the potential advantages and limitations of geopolymer waste forms, providing a critical evaluation of current research findings on radioactive waste disposal. Geopolymer waste forms offer superior stability and sustainability, potentially revolutionizing nuclear waste management practices.

Keywords: Radioactive waste, Immobilization, Geopolymer, Waste form

A Study on Re-Solidification of Inner Waste in Large Concrete Drums

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In early stage of Kori unit 1 nuclear power plant (NPP), boron concentrates were solidified using cement and packaged in 200 L drums. The large concrete drums were applied to ensure the safe storage and reduction of worker radiation exposure. The inner cement solidified drums require additional treatment to satisfy the waste acceptance criteria (WAC) of disposal facility. The disposal of solidified form requires various properties as shown in WAC. The WAC identifies various properties, including mechanical property, radiation resistance, chemical stability, etc. Since the cement solidified form alone does not satisfy the current WAC, addition of adequate material and process is required. The candidate materials are cement, geopolymers, organic polymers, etc. The crush process of cement solidified form is also required to enable the re-solidification process.

In this study, re-solidification of crushed cement solidified form was systematically studied. The candidate materials for re-solidification are geopolymers, glass matrix, cement, etc.

Keywords: Solidification, Geopolymer, Vitrification, WAC

Development of Over-Packaging Container for Polyethylene High Integrity Container (PE-HIC) in Korea

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Polyethylene High Integrity Containers (PE-HIC) have been used to store waste resin generated in domestic nuclear power plants. However, because of structural stability and shielding requirement for disposal, it is necessary to study the disposal suitability of PE-HIC. In particular, the domestic transportation requirement and disposal acceptance criteria must satisfy the surface dose rate of 2 mSv/hr. Thus the development of over-packaging container is required due to the high dose of waste resin. Through the development of over-packaging container to satisfy the structural stability and shielding requirement of disposal site, it will be reduced radiation exposure of workers due to transportation or disposal of PE-HIC, and meet the requirements for acceptance criteria of the disposal site.

Keywords: Over-packaging, PE-HIC

Structural Evaluation of Improved Disposal Canister Emplacement and Retrieval Device Under Lifting Condition

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The Deep Borehole Disposal (DBD) method has various advantages, such as minimizing the surface area of the disposal site, reducing corrosion of the disposal canister, and enhancing long-term structural safety. However, it is essential to address challenges related to the emplacement, retrieval of the disposal canister, and sealing of the borehole. In previous designs, the disposal canister connecting device was deployed and retrieved as a single unit, leading to potential issues such as poor canister engagement and device malfunction. Hence, this study aimed to design an improved emplacement and retrieval connecting device to mitigate these challenges. The purpose of this study was to evaluate the structural integrity of an improved emplacement and retrieval device of a DBD container. The disposal canister connecting device was evaluated according to ANSI 14.6 standards. The allowable stress should be less than the yield strength under the load condition of 3 g. The disposal canister connecting device for emplacement is approximately 1,645 mm long, 406 mm in diameter, and weighs 688 kg. The disposal canister connecting device for retrieval is approximately 1,325 mm long, 406 mm in diameter, and weighs 464 kg. The disposal canister connecting device can handle up to 10 disposal canisters, each weighing up to 2.2 tons, and is bolted to the canister. The disposal canister connecting device was made of SUS 304 stainless steel and the maximum operating temperature was about 150°C. For structural evaluation, ABAQUS finite element analysis program was used. The analysis model was modeled only 1/2 part considering symmetry condition. The emplacement device model comprised 94,606 nodes and 76,077 solid elements, while the retrieval device model consisted of 67,369 nodes and 53,375 solid elements. The canister was modeled as a dummy with the equivalent weight. Axisymmetric conditions were applied to the symmetrical surface of the disposal canister, and vertical restraints were applied to the upper lifting lugs. A surface-to-surface contact condition was applied to the contact part, with a coefficient of friction of 0.2, typical for steel.

The analysis revealed that the highest stress occurred in the lifting hook directly lifting the canister connector, reaching 145.4 MPa in the emplacement device and 144.4 MPa in the retrieval device. Although the lifting plates satisfied the design stress, the safety factor is relatively low at 1.06, suggesting a need for improved yield strength of material or modification of geometry is required for further detailed design refinement.

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Keywords: Structural integrity, Deep borehole disposal canister emplacement and retrieval device, Lifting conditions, Deep borehole disposal (DBD) method

Review on the Design of Cold Crucible Induction Melter for Radioactive Wastes

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Various treatment technologies for disposal of radioactive waste at home and abroad have been studied for a long time. The developed technology can be largely divided into high temperature treatment technologies such as incinerators and vitrification and low temperature treatment technologies such as cement solidification, among which vitrification technologies belonging to high temperature treatment technologies were developed for the treatment of Low and Intermediate Level Radioactive Waste (LILW) as well as High Level Radioactive Waste (HLW). The glass melting melter used in vitrification technology uses Joule Heated Melter (JHM), a method of heating using metal electrodes inside the glass melting melter, and the Cold Crucible Induction Melter (CCIM), a method of cooling the glass melting melter metal sector with cooling water and melting the glass inside with an induction coil installed outside. The CCIM was developed for the treatment of LILW in South Korea because it has the advantage of being smaller in size than JHM. Overseas, the CCIM was developed to replace the JHM, which vitrifies HLW. In this study, the design of the CCIM developed at home and abroad was analyzed.

Keywords: CCIM, Radioactive waste

Management of Radioactive Solid Waste From the HANARO in 2023

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The HANARO is multipurpose research reactor that has been operated since 1995. The reactor core of HANARO is operated under controlled temperature and pressure conditions, with nuclear fuels undergoing approximately six months before replacement. In case of fuel damage during the reactor operation, the radiation monitors located at the upper part of the reactor coolant system and the reactor water pool promptly detect radiation leading to reactor shutdown.

The design basis for the radioactive solid wastes generated from the reactor is to transfer the wastes to the in-house radioactive waste management facility safely. The radioactive solid waste management system is composed of a transport canister that contains filters, waste papers, etc., and a crane of the building. However, no specialized system is designated for the treatment of radioactive solid waste. In 2023, The amount of radioactive solid waste generated from the reactor building was 1.0 m^3 , the amount of transport was 0.4 m^3 , and the total amount of storage was 6.6 m^3 .

The low-level radioactive solid wastes are collected in vinyl bags, and then packed into drums for transportation. The medium-level radioactive solid wastes such as filters, resins from desalting equipment undergo drying, packing into a transport canister, and transported to the in-house facility. The exposure rate of the radioactive waste transport canister outside is lower than 2 mSv/hr .

Before transferring the radioactive solid waste to a radioactive waste treatment facility, contamination inspection of the outer surface of the container, dose measurement, and external dose indication of the container are carried out in accordance with the institute's radioactive waste treatment procedures. This paper describes the treatment status of radioactive solid waste generated in the HANARO Reactor building in 2023.

Keywords: HANARO, Radioactive solid waste, Exposure rate, Transport canister

Review on the Validation of Activated Material Modeling Analysis

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Radioactivity analysis evaluation is necessary for the disposal of metallic waste generated from nuclear power plant. Currently, in the case of domestic nuclear power plants, a preliminary evaluation is performed, and a radwaste treatment plan is established according to the results. After radwaste treatment, waste classification and disposal are performed through radioactivity analysis using representative samples.

Overseas, radioactivity analysis evaluation is conducted in advance for radioactive waste disposal through radioactivity modeling and scaling factor, which are non-destructive techniques. Through this, the level of waste is classified and the effectiveness of the waste evaluation is verified before disposal.

Currently, various studies are being conducted in the EU to verify the effectiveness of modeling. In this study, we analyzed the performance details of MCNP6/SCALE6.2, MCNP+/ORIGEN-S and ADVANTG, which are used to verify radioactivity modeling. In addition, a study to verify the effectiveness of the scaling factor was reviewed by comparing actual waste from BWR and RBMK-1500 nuclear power plants with a destructive analysis method.

In the future, we plan to secure a method to verify the effectiveness of domestic evaluation through additional radioactivity analysis modeling verification and scaling factor verification research.

Keywords: Radioactivity analysis, Scaling factor, Non-destructive evaluation

Review on the Indirect Evaluation Methodology for the Radionuclide Inventory of the Dry Active Waste (DAW) Drums Generated in the New NPPs

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The radionuclide inventory of dry active waste (DAW) drums is evaluated through the direct measure by radwaste drum assay system (RAS) with the indirect evaluation by a scaling factor (SF) of the difficult-to-measure (DTM) nuclide against easy-to-measure nuclide (ETM) nuclide that is Co-60 or Cs-137. The DTM and ETM nuclides in a representative sample of DAW should be quantified reliably above minimum detectable activity (MDA) for the determination of scaling factors.

However, the nuclide concentration of the dry active waste generated in the new nuclear power plants (NPPs) generally tends to be lower than old NPPs because of technical advancements in the reactor core management, nuclear fuel design and manufacturing, plant operation.

We reviewed on the reasonable methodology based on efficiency that can evaluate indirectly the specific radioactivity of DTM nuclides with the scaling factor, mean activity concentration method and advanced nuclide analysis technology and manage continuously the effectiveness of the indirect evaluation methods.

Keywords: New NPPs, Scaling factor, Mean activity concentration, Indirect evaluation

Review of Latest Radioactive Waste Monitoring Technology

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Due to the development of the latest IT technology, research is underway on the applicability of various technologies in the field of radioactive waste. Research using the latest sensor technology and deep learning technology is being conducted not only domestically but also overseas. We plan to select technologies that can be applied domestically through analysis of recent technology development trends.

Some of the representative technologies are introduced as follows. RFID sensors are embedded in waste packaging containers and used to acquire information such as temperature, pressure, humidity, pH value, O₂ and N₂ concentration, and radiation dose. In order to acquire these various parameters, research is being conducted overseas to develop sensors that can minimize battery life and operate without long-term maintenance. In addition, research is being conducted on using optical sensors to wirelessly detect temperature, pressure, humidity, and radiation distribution of radwaste on the surface or inside for a long period of time. Lastly, research on surveillance technology using digital twins using machine learning algorithms and neural networks was also being actively conducted.

We plan to conduct research on the applicability of these latest technologies in domestic nuclear power plant through selection and research on their feasibility in the field of radwaste.

Keywords: IT, Radwaste, Monitoring, Sensor

Status Update of the Progress of the Hanford WTP Vitrification in US DOE Hanford Site

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Hanford site has been operated since 1943 to produce the by-product of plutonium for nuclear bomb. Significant amount of radioactive wastes was generated from the nuclear weapons production process. The by-product is being stored in a place called tank farm of Hanford site. There are 177 old tanks buried in the ground in this area. Due to the risk of leakage into the air and the Columbia River, the US DOE and EPA, and Washington State Department of Ecology organized the Tri-Party Agreement (TPA) to clean-up the Hanford site in 1989. To treat significant amount of radioactive wastes, the WTP (Waste Treatment Plant) to vitrify about 212 million liters of radioactive waste has been built and is currently conducting a cold-test for commercial operation for low-activity waste facility.

Keywords: Waste treatment plant, Vitrification, Hanford site, Commercial operation, Cold-test

Status Update of LLW Disposal Facilities and PE-HIC Disposal for Radioactive Wastes in the United States

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In the United States, the disposal of LLW from nuclear power plants is carried out at four facilities, Richland, Clive, Barnwell and Andrews. The Richland disposal facility is located north of Richland of Washington state and has been operated by US Ecology since 1965. The facility is one of only three full-service Class A, B and C LLW disposal facilities. The Clive disposal facility is located in the West Desert of Utah and owned by EnergySolutions and licensed to dispose of Class A LLW. The Barnwell disposal facility is owned by the state of South Carolina and operated by EnergySolutions. The facility has been operated since 1971 and is the host disposal site for the Atlantic Compact which is comprised of SC, NJ, and CT. The facility is licensed to dispose of Class A, B and C LLW, including irradiated steam generators, resins, and reactor pressure vessels. The Andrews disposal facility in Andrews County, Texas, is owned by Waste Control Specialists (WCS) and is licensed to dispose of Class A, B and C LIW.

Keywords: Richland, Clive, Barnwell, Andrews

Recent Status for Contaminated Water Management Generated From Fukushima Daiichi NPP

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On 24 August 2023, Japan started releasing the contaminated water stored from the tanks of the Fukushima Daiichi nuclear power plant site into the sea. Tokyo Electric Power Company discharged about 23,351 tons of contaminated water through three discharges, and when the fourth discharge is completed, the total amount will be about 31,200 tons. Also, Japan's government plans to discharge 54,600 tons of contaminated water from April this year to March next year. The contaminated water is treated by advanced liquid processing system (ALPS) to remove 62 radionuclide containing Cesium, Strontium, Iodine and etc. using co-precipitation (or precipitation) and adsorption for other nuclides (except for Tritium and Carbon-14). The total amount of the contaminated water generated by ALPS facility is 1,325,969 m³ (as of March 21, 2024).

Keywords: Contaminated water, Fukushima daiichi NPP, Discharge, ALPS

Japanese Practice of Assessing Radioactive Inventory of Radwaste Focusing on Mean Activity Concentration Method

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The assessment of radioactive inventory is crucial for the disposal of radwaste. In Korea, the radioactive inventory of radwaste from nuclear power plants is typically evaluated using an indirect method known as the scaling factor (SF) method. Each SF value for a specific difficult-to-measure (DTM) nuclide indicates its concentration ratio to the corresponding gamma-emitting nuclide. Consequently, the concentration of a DTM in a particular radwaste package is estimated by multiplying its SF by the concentration of the corresponding gamma-emitting nuclide measured for the package.

The SF method operates on the premise of a significant concentration correlation between the DTM nuclide and its gamma-emitting counterpart, suggesting a proportional relationship between their concentrations. However, instances have been reported where such a correlation cannot be established for various reasons. In such cases, the International Organization for Standardization (ISO) and the International Atomic Energy Agency (IAEA) recommend using the mean activity concentration method. This method assumes the arithmetic mean activity concentration of a specific DTM nuclide, derived from several representative radwaste packages, to represent the concentration of the DTM in a population of packages from which the representatives were selected, independently of the concentration of the corresponding gamma-emitting nuclide.

This study presents the Japanese approach to assessing radioactive inventory. Japanese regulatory guidelines permit not only the use of the SF method but also the mean activity concentration method. Japanese utilities including nuclear power plant operators have adopted the SF method for assessing the radioactive inventory of radwaste generated from both operating nuclear power plants and decommissioned nuclear facilities. The Japanese experience is anticipated to offer valuable insights to the Korean nuclear power generation industry when dealing with radwaste scenarios where the scaling factor method proves challenging to apply.

Keywords: Radwaste, Difficult-to-measure nuclide, Radioactive inventory, Mean activity concentration, Scaling factor

A Case Study on Activation Calculation of Non-Fuel Materials in the SFP of PWR

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There are various types of non-fuel material irradiated by being depleted fuel in the core or by spent fuel in the SFP (Spent Fuel Pool) in the SFP of domestic PWR (Pressurized Water Reactor). These materials include fuel skeleton from which nuclear fuel rods have been removed due to damage, control rods with expired performance and design life, burnable poison rods, thimble plugs, neutron sources, in-core instruments (ICIs), and dummy fuels. In addition, Trash Basket, which stores these non-fuel materials, is also a non-fuel material. According to the Nuclear Safety and Security Commission Notice No. 2020-6, for radioactive waste disposal, it is necessary to classify what level the material falls into. In order to confirm the specific radiation level of non-fuel material, there are two methods, indirect method such as computer simulation and direct method such as sample analysis. In this paper, non-fuel materials are evaluated using computer simulation to check how much they were activated. If we know how much exposure has been in such an environment, the degree of irradiation can be evaluated using MCNP, ORIGEN, FISPACT-II, etc. Typically, the activation calculation proceeds in two stages, Step 1 evaluates the neutron flux in the region where the objects to be evaluated are operated. Step 2 evaluates the degree of activation of the objects. Non-fuel materials such as ICI, neutron sources, and burnable poison rods operated in the fuel zone can use neutron flux in the fuel zone, but non-fuel materials such as control rod, thimble plug operated are evaluated considering the neutron flux ratio of the fuel top zone and fuel zone because flux of fuel zone can produce excessive activation results. For non-fuel materials that have never been irradiated in the core, they are evaluated assuming that they were operated in the spent fuel zone in the SFP. Of course, ICIs, Control rods, thimble plugs and burnable poison rods were also evaluated considering the period that have been irradiated in the core and SFP. Most of non-fuel materials were evaluated as intermediate level radioactive waste but because there is conservative assumption such as neutron exposure time and excessive neutron flux level in the SFP area, it is judged that the evaluation results have conservatism. This is the result of the preliminary study. In the future, additional evaluation will be conducted for best estimate when long term research is launched according to KHNP technical development road map.

Keywords: Non-fuel material, Control rod disposal, ICI disposal, Radioactive waste, Activation calculation

A Case Study on Nuclide Analysis of Non-Fuel Materials in the SFP of PWR

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According to the Radioactive Waste Management Act, when radioactive waste of limited types and quantities occurs under Nuclear Safety Act, the radioactive waste generator shall deliver it to the radioactive waste management business operator. To request the operator of a radioactive waste disposal facility to take over the waste, various information for waste management shall be presented. Radionuclides constituting more than 95% of the total amount of radiation contained in the waste shall be identified, and the concentration of 13 additional radionuclides including H-3, C-14, Fe-55, etc. and gross alpha shall be identified. I-129, H-3, C-14 are classified as volatile nuclides and radioactive concentration can be measured using the volatility of these nuclides. Other nuclides are classified as non-volatile nuclides, of which the radioactivity of gamma emitting nuclides (Ce-144, Cs-137, Co-58, Co-60, and Nb-94) can be measured without chemical separation using a solutioned sample. The remaining Fe-55, Ni-59, Ni-63, Sr-90, and Tc-99 are classified as difficult to measure radionuclides and measured individually after going through a chemical separation process in the sample solution. As described above, various conditions are involved in delivering it to a radioactive waste disposal business operator for the disposal of radioactive waste. Among them, this paper intends to describe a case study on the radionuclide identification of non-fuel material in the SFP. These materials include fuel skeleton from which nuclear fuel rods have been removed due to damage, control rods with expired performance and design life, burnable poison rods, thimble plugs, neutron sources, in-core instruments (ICIs), and dummy fuels. These materials consist of B₄C, Inconel-625, Al₂O₃, Inconel-600, Rh, Cf-252, Sb-Be, SS304, Zircaloy-4, Zirlo by each material. According to the results of the activation calculation through computer simulation, each non-fuel material includes H-3, Ni-63, Co-60, Nb-93m, Ni-59, Nb-94, Ag-108m, C-14, Mo-93, Fe-55, Al-28, Rh-102, Rh-102M, Rh-104 which are 13 mandatory identified radionuclides and other radionuclides. Therefore, it should be considered that the radiochemical analysis for Zr-93, Nb-93m, Mo-93, Cd-113m are conducted to deliver non-fuel materials in the SFP. However, considering the proportion of other nuclides in the total amount, It seems that the necessity for analysis of other nuclides is more likely to be determined after consultation with regulatory body and radioactive waste management business operator.

Keywords: Non-fuel material, Mandatory identified radionuclide, Difficult to measure radionuclide, Radiochemical analysis

Preliminary Study of Radiation Shielding Analysis for ILW Hot Cell Workers

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Intermediate-level waste (ILW) is typically generated from commercial nuclear power plants as a result of neutron activation on reactor structures. According to waste acceptance criteria (WAC), more than 95% of the radionuclides in the radioactive waste must be identified, making characterization of radioactive waste essential for proper radioactive waste disposal. However, characterizing of ILW can pose risk to workers due to its radioactive properties. Thus, waste characterization must be conducted in a radiation-shielded facility, such as hot cell.

Although the hot cell is typically designed for handling high-level waste (HLW) or nuclear fuel, it is also necessary for ILW to ensure worker safety during analysis. Within the ILW hot cell, various experiments will be conducted to identify radionuclides in the waste before disposal. It is necessary to evaluate whether the radiation received by workers during operations remains below the domestic effective dose limit. To calculate the dose rate to workers, the Monte Carlo simulation code, MCNP, was utilized for radiation shielding analysis in this paper. To ensure the radiological safety of workers, it is required to show whether the dose rate of workers remains below the effective dose limit in Korea.

In this paper, a 3D model of ILW hot cell has been simply designed and the effective dose rate to workers was evaluated by placing sample of ILW source within the hot cell. The ILW hot cell was designed as simple as possible with two lead-shielded window and a hatch door installed on the roof of the room. The detection points of dose rate are located at 10 cm away from the lead-shielded window, and 10 cm away from the hatch door. In addition, a simple waste management scenario in hot cell was derived for evaluating the work time in operation area. Therefore, the position of radiation source in hot cell would move along the waste management scenario. As a result of the radiation shielding analysis for workers, the ILW hot cell facility could satisfy the effective dose limit with the dose conversion factor of ICRP-72. For future work, it is planned to design specific 3D model of the facility with detailed hot cell waste management scenario which include physical and chemical experiments.

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Keywords: Intermediate level waste, Monte carlo, Radiation shielding, Radioactive waste characterization, Effective dose limit

Development of ODCM and Its Application to Operational Dose Assessment of Gyeongju LILW Repository

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KORAD has operated Gyeongju LILW repository since after 2014 when operation of the repository was finally approved by Korean regulatory body NSSC. As one of the most important requirements of the operation, KORAD should carry out off-site dose calculation in association with comprehensive site surveillance and radiation monitoring. In that regard, KORAD developed its own off-site dose calculation manual (ODCM) mainly consisting of RADCAP coupled with meteorological analysis module and applied the calculation manual to obtain the operational dose in and out of the disposal facility. These activities are subject to regular reporting to the regulatory body for a multiple check for safe operation of the repository in terms of radiological environment safety. As of the end of 2023, since 2015 KORAD has received approximately 30,000 radioactive waste packages from waste generators and disposed most of the introduced packages while the rest of the packages are being stored in storage building waiting for the final disposal.

This paper has an intention to show summarizing result from ODCM starting from 2015 to 2023 up to date. It also shows a trend of dose variation as the number of introduced radioactive waste package being increased. Trend of calculated radiation level over the operation period of time has not been changed significantly because current radioactive wastes being disposed or stored are not high enough to cause high radiation dose level and variation to the repository and its environment. In this paper we shortly introduced how to prepare the ODCM and to conduct the calculation and finally show possible variation in the calculation result over the operation time.

Keywords: ODCM, Environment safety, Radioactive waste, Regulatory requirement

Calculation of Radioactive Sensitivity According to Irradiated Stainless Steel-304 Composition Tolerance

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Radioactive waste is defined as waste that contains radionuclides exceeding regulated concentration levels or contaminated with radionuclides. In South Korea, the majority of radioactive waste comes from nuclear power plants. With the decommissioning of nuclear reactors, a diverse and large amount of radioactive waste is expected to be generated, which requires preliminary classification for safe management of these wastes.

The purpose of this study is to perform an activation analysis and estimate the level of radioactive waste in structures composed of SS304, specifically the Former and Baffle, within the OPR-1000, a standard South Korean nuclear power plant. SS304 is primarily composed of iron, chromium, and nickel, with minor elements such as manganese and silicon. Variations in the composition tolerance of these elements can change the neutron absorption cross section of the material, significantly impacting the results of the activation analysis. This paper investigates these effects by performing an activation analysis on 13 specified compositions of SS304, as listed in NUREG-3474, using the ORIGEN-S code.

To analyze composition tolerance for SS304, a neutron flux of $3.57\text{E}+13 \text{ \#}/\text{cm}^2\cdot\text{s}$ was used, referencing the activation analysis paper for OPR-1000. The analysis conducted with this neutron flux classified all 13 SS304 compositions as intermediate-level waste. However, it was observed that the inventory of the radionuclide nb-94 varied significantly with composition tolerance. Therefore, when classifying radioactive waste, it is crucial to consider the composition tolerance according to the manufacturer.

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Keywords: Activation analysis, SS304, Composition tolerance

A Review on International Radioactive Waste Management and Disposal Facilities

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It is generally accepted that the nuclear fuel cycle generates radioactive waste in a variety of forms, characteristics and radioactive levels. The radioactive wastes are classified in to into different categories, based on their hazard level and its duration. Over time, the radioactivity of the radioactive waste will decrease naturally as the result of natural decay and eventually they are converted into stable form. The half-life of the specific radioactive nuclide defines the above characteristics. Since the disposal conditions are different in many countries, the waste classification and safety analysis methods are different in many countries. For example, a surface or near-surface disposal facility, which is a general disposal structure in many countries, is generally designed around a hazardous life-time for the waste of ~300 years (the equivalent to about ten half-lives of Cs-137).

In this study, systematic review on ONKALO disposal facility will be reported, including history of Finland waste management history, multi-barrier system and structure, encapsulation plant, etc.

Keywords: Spent fuel, Onkalo, Disposal, Posiva

A Review on Radioactive Waste Management of Japan

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Japanese radioactive waste classification system categorized radioactive wastes into high level waste and low-level waste. There are relative high level waste (L1), relatively low level waste (L2), and very low level waste (L3) in the system. The examples of L1 are of control rods, reactor vessel internals, solidified liquid waste, etc. The examples of L2 are various spent equipment. The examples of L3 are concrete, metal, etc. The radioactive waste, generated during operation of nuclear power plants, are transported and disposed in Rokkasho Low-Level Radioactive Waste Disposal Center. The accumulated disposed radioactive waste is ~350,000 drums as of 2023. The spent fuel, generated in Japanese nuclear power plant, had been reprocessed in UK and France. The vitrified forms from reprocessing were sent back to Japan. They were stored in Vitrified Waste Storage Center.

In this study, systematic review on radioactive waste management of Japan will be discussed, including disposal system, current status, vitrification process, vitrified form disposal status, etc.

Keywords: Vitrification, Japan, Rokkasho, Radioactive waste

A Preliminary Study on Improving Classification and Management Methods for Radioisotope Waste in Korea

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There has been an increase in facilities using Radioisotopes for various purposes such as medical, industrial, and research activities in Korea, leading to a continuous increase in the generation of Radioisotope (RI) waste. However, most of RI waste is not being accepted to disposal facility. The issue arises from the generation characteristics of the RI waste. Since RI wastes are used for various purposes across different fields, and its characteristics are complex and diverse. Consequently, there has been a lack of specific guidelines for the classification, waste acceptance criteria (WAC), and disposal methods for RI waste.

There is a growing need to present a revised classification and management plan that specifies the WAC for RI waste, to facilitate the smooth transfer of waste and its efficient management. This study has investigated and compared the domestic and international classification systems and IAEA recommendations to derive a management system for RI wastes. The radioactive classification system and management methods in France and Sweden were investigated and analyzed for applicability to domestic RI waste. The D-Value used in the classification of radioisotopes in IAEA RS-G-1.9 was examined to see if it could be applied to the classification of RI waste in Korea, and a management plan was derived.

The proposed improvement plan aims to maximize the amount of waste that can be recycled, clearance, or stored until decay to a manageable level for clearance. This approach is expected to increase flexibility in waste handling and allow disposal facility licensee to focus more efficiently on managing higher-risk waste categories.

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Keywords: Radioisotope, RI waste, Classification, Management, D-Value

Directions to Improve Public Acceptance in a Host City of High Level Waste Disposal Facility–Final Repository

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This paper is to provide right directions to improve public acceptance in a host city of a high level waste disposal facility which is a final repository in Korea on the basis of the best practices to host the low and intermediate level radioactive waste (LILW) management facility in Gyeong-ju, Korea.

After the host city selected in 2005, the central government made promises to perform two different support projects, which were 4 special support projects and 55 general support projects with the local government of the host city, Gyeong-ju.

Four special support projects were KORAD headquarter relocation from Yong-in to the host city, Multi-purpose accelerator complex establishment, around \$300 million fund from the central government to the local government in the host city and subsidies when LILW deliver to the disposal facility.

The Gyeong-ju municipality applied for 116 general support projects in 2006 (around \$8.8 billion USD) as the host city which was selected in the LILW disposal facility. However, the central government accepted 55 general support projects out of 116 in 2007 which was around \$3.2 billion. The selected projects include ways to improve public acceptance, to boost local economy, to promote tourism industry, to provide electricity bills for local residents, to bring workers back from big cities, to build industry complex and to perform many other projects.

In this paper, I will present the local resident expectations of support projects with the initial plan and operation methods in a host city for the high level radioactive waste disposal facility in Korea. Then I will give you reasons that why we are required to have a master plan and stepwise action plans for the support project in the host city. Followed by the provided reasons, flexible operation are necessary on the basis of environment change and local residents requests. Moreover, organized government budget should be accompanied separately from the local government budget. The last is that integrated governance structure should be established with local resident participation in the support projects.

To wrap up, a host city is helpful to bring more citizens and boost local economy with the proposals that I discussed in the paper.

Keywords: Public acceptance, Low and intermediate level radioactive waste, High level waste, Disposal facility, Support project

Introduction to Development and Commercial Operation of Vitrification Plant China (VPC) for HLW Treatment

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On September 11, 2021, a high-level radioactive waste (HLW) vitrification commercial facility (Vitrification Plant China) located in Guangyuan, Sichuan, China, began commercial operation. Radioactive waste to be treated at the vitrification facility is HLW generated during reprocessing of spent fuel as part of the defense project, and is about 800 m³. The core facility of the vitrification facility was introduced by a consortium of German industries and a ceramic melter with Inconel 690 electrode inserted was used. China plans to reprocess fuel generated from commercial nuclear power plants and rationalize the generated HLW. HLW is scheduled to be disposed of at a deep geological repository, and the Beishan disposal site in Gansu province is being prepared as a likely disposal site.

Keywords: High-level radioactive waste, Vitrification plant china, Ceramic melter, Beishan disposal site

Preliminary Disposal Cost Assessment According to Spent Activated Carbon Processing Method

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Spent activated carbon (SAC) discharged from nuclear facilities is typically classified as very low-level radioactive waste (VLLW) and is disposed of as ordinary industrial waste after its radioactivity naturally decays to disposal levels following on-site storage. However, due to the presence of long-lived radioactive isotopes such as C-14, the disposal classification for this waste shifts from self-disposal to VLLW. Therefore, the disposal method of spent resin can be applied to the case of SAC disposal.

Two processing methods for spent resin are available in Korea: one is to solidify it with cement and dispose of it in 200 L drums, while the other is to dispose of it in a High Integrity Container (HIC) without solidification. An HIC is a waste packaging container capable of maintaining structural integrity for over 300 years under typical Korean underground conditions and disposal criteria. Developed HIC are designed in two sizes: 510 L and 860 L. The volume of waste disposed of in an HIC depends on the particle size of the solid waste being disposed of.

In this paper, a disposal cost assessment of SAC was performed through cement solidification in 200 L drum and disposal in 860L HIC. For the cost assessment, the total volume of SAC was assumed to be 1,000,000 L. Since the lack of the actual content rate about solidified with cement, content rates have been assumed to be increased from 10% to 90% with 10% of interval. Additionally, for the case of HIC, filling rates of the SAC were assumed to be 95%, 90%, ... 20% due to the porosity of SAC. Thus, total 9 cases of each processing method have been proposed in this paper to perform cost assessment. Factors considered for economic evaluation include “content rate”, “filling rate”, “volume of disposal container”, “number of required drums”, “drum unit cost”, and “total disposal cost”.

As a result, it was found that the cost of each processing method has decreased or increased exponentially. In addition, the cost of each processing method has shown similar in the case of content rate with 40% with 200 L drum and filling rate of 70% with 860 L HIC. However, the cost of 200 L drum has shown 6.3 billion won higher than the cost of 860 L HIC. Further analysis would be performed to determine the detailed cost assessment for SAC disposal.

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Keywords: Activated carbon, Cost assessment, Disposal, Radioactive waste, HIC

Inspection of Radioactive Waste Through 4th Industrial Revolution Technology

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Currently, the radioactive waste (radwaste) in the disposal facility is disposed of after confirming the disposal safety of the waste by inspecting radwaste from generator and disposal facility. There are various inspection items to confirm the safety of the waste. Among them, two items, which inspect the presence of harmful substances that promote explosion or nuclide leakage and radioactivity in the waste by analyzing the concentration of radionuclides, are discussed the most importantly now. There are a lot of data to support these inspection items, so the inspection takes long time and a lot of data cause human errors during inspection. Therefore, we need technology to efficiently perform waste inspection, reduce human errors, and manage a lot of data collectively. So among the 4th Industrial Revolution Technologies, we will introduce Deep Learning Technology & Big Data Computerization for efficient inspection and Waste Tracking System (WTS) for comprehensive data management.

Depending on the type of contents in radwaste, nuclide leakage, explosion or toxicity may cause. When packaging radwaste, four photos (25%, 50%, 75%, 100% stage height) are taken and recorded as data to prove presence of harmful substances. However, because of large quantity of radwaste, the inspection time takes a long time, and if inspector makes a judgment based on the photo, the standard may not be consistent. So we use AI Judgement Technology. Based on Big Data obtained from previous inspections, AI trains the technology to classify contents in a lot of photos through Deep Learning Technology, and judges many photos in a short time during inspection, increasing reliability due to efficient and consistent standards.

Also there is extensive data from analysis of each nuclide's radioactivity concentration. Based on this data, the amount of radioactivity is calculated and the safety of disposal in the disposal facility is evaluated. In other words, computerization is necessary to calculate nuclide concentration data uniformly. If the data, derived from the HPGe Gamma Spectrometer in Pdf file originally, is converted to Excel file and automatically calculated, human errors and the time to inspect the data can be greatly reduced.

Furthermore, the WTS can efficiently manage the history of radwaste by tracking it through GPS from generation to inspection, transportation, and disposal. In addition, the data, derived from radwaste inspection, is stored in the system. So information on radwaste can be checked at once and can be permanently managed after disposal, also improving safety.

Keywords: Waste tracking system (WTS), Deep learning technology, Big data computerization, GPS

Preliminary Consideration of Safety Assessment for a Landfill Disposal Facility Using RESRAD-ONSITE Code

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In Korea, Kori Unit 1 and Wolsong Unit 1 have been permanently shut down with decommissioning plans in progress. More nuclear power plants are expected to go through the same process after continued operation. During the decommissioning of nuclear power plants, a large amount of radioactive wastes with various characteristics are generated. Therefore, safe management of radioactive wastes is one of the significant aspects of decommissioning. Radioactive wastes classified as very low-level waste make up a large proportion. Prior research concerning the expected amount of radioactive wastes from the decommissioning of Kori Unit 1 has shown that very low-level wastes would account for about 54.6% of the entire radioactive waste.

For the disposal of very low-level wastes, countries such as the United States, Sweden, and France have been operating landfill disposal facilities. In Korea, preparation for the construction of the landfill disposal facility is undergoing. Although very low-level wastes can be disposed into the underground silo or the near-surface disposal facility currently existing or under construction in Korea, landfill disposal has advantages such as cost reduction and capacity save of the other facilities for the disposal of low-level wastes or intermediate-level wastes.

In order to operate the landfill disposal facility with an operating license, safety assessments for the workers and the public are necessary. In this study, total radiation dose and long-term radiological risks to human health and the environment were evaluated based on scenarios and exposure pathways, as well as major factors affecting were found using the RESRAD-ONSITE code.

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Keywords: Landfill disposal, Very low-level waste, Safety assessment, RESRAD-ONSITE

A Systematic Approach to Improving Regulations to Manage Mixed Waste Efficiently

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The management of mixed waste, radioactive waste containing hazardous materials, presents regulatory challenges due to the lack of specific regulations in domestic disposal facility acceptance standards. While hazardous substances are regulated based on chemical characteristics, the current criteria do not address the complexities of mixed waste adequately. This paper examines the regulatory landscape and proposes a systematic improvement plan to enhance acceptance criteria for hazardous waste generated from domestic nuclear power plants.

Despite the regulation of hazardous substances within acceptance standards, the focus remains on chemical characteristics, leaving a gap in addressing radioactive waste. To deliver radioactive waste containing hazardous materials to a disposal facility, the risk must be removed by decomposing the substance to 0.1 to 1% or less. However, limitations exist in completely decomposing hazardous substances within radioactive materials, without guidance for cases where removal is not feasible.

To address these challenges, the United States established the Final Rule for Mixed Waste in 2001, providing exemptions from certain RCRA regulations. RCRA classifies hazardous substances based on characteristics and lists, offering a framework for evaluating hazardous waste. However, the dual regulation of low-level mixed waste necessitates a more comprehensive approach.

This paper proposes a systematic improvement plan focusing on enhancing acceptance criteria for hazardous waste from domestic nuclear power plants. By studying the characteristics of various mixed wastes, compliance with both the Atomic Energy Act and the Environmental Act can be ensured. The plan identifies specific hazardous wastes requiring urgent disposal, such as asbestos, waste batteries, hazardous cans, lead, and aluminum, and presents acquisition criteria for their management.

In conclusion, addressing regulatory challenges in managing hazardous mixed waste requires a systematic approach. By enhancing acceptance criteria and ensuring compliance with relevant legislation, we can effectively manage hazardous waste from domestic nuclear power plants, minimizing environmental and public health risks.

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

Proposal of Concept for On-Site Disposal of Decommissioning Waste From NPPs

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Kori Unit 1 and Wolsong Unit 1 have been permanently shut down and are in the process of decommissioning. The decommissioning of nuclear power plants is expected to generate a significant amount of decommissioning waste. Based on prior research, it is estimated that approximately 80,000 drums of decommissioning waste, classified as radioactive waste, will be produced, using Kori Unit 1 as a reference. Given the target disposal rate of 14,500 drums per nuclear power plant (NPP), extensive decontamination procedures are anticipated. There are 30 nuclear power plant units in the Republic of Korea, including those permanently shut down, operational, or under construction. With all nuclear power plants slated for permanent shutdown and decommissioning in the future, the total volume of decommissioning waste is projected to reach around 435,000 drums, exceeding the current capacity of the radioactive waste repository in the Republic of Korea, which stands at 385,000 drums. Furthermore, considering the radioactive waste generated during operation, there are concerns about the capacity of the radioactive waste repository.

Some countries, such as the United States, Canada, Germany, and Finland, have conducted on-site disposal or conducted related studies as an alternative to utilizing the radioactive waste repository for decommissioning waste disposal. On-site disposal offers advantages such as cost reduction, minimized public interaction due to reduced off-site waste transportation, and decreased occupational radiation exposure. However, drawbacks include challenges with public acceptance, restrictions on the disposal of only short-lived radionuclides, and the need to increase the number of radioactive waste repositories within the country. A study conducted in Germany focused on the effects and technical feasibility of on-site sinking (i.e., on-site disposal) of the reactor building, confirming the absence of radionuclide release into the environment over an extended period.

To address the limitations of on-site disposal, this study proposes disposing of radioactive waste generated during the decommissioning of multiple nuclear power plants at the reactor building of one NPP located on the same site. It is expected that the disposal of radioactive waste including long-lived radionuclides at a reactor building, which has high integrity, can be accomplished, as demonstrated in previous studies. To validate the applicability of this method, a safety assessment was conducted using the AMBER software.

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Keywords: On-site disposal, Safety assessment, Decommissioning, Radioactive waste, AMBER

Improving Safety and Efficiency of Radioactive Waste Disposal Through Coating Application

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Radioactive waste is disposed of using 200 and 320-liter disposal containers at a radioactive waste repository. During transportation to repository or during storage, disposal containers may be damaged, potentially compromising their ability to maintain radioactivity safety. Various studies are being conducted to enhance the safety of disposal containers, and in this study, the application of coatings is considered to improve the structural/containment performance of disposal containers.

In this study, tungsten carbide (WC), capable of shielding gamma rays, and boron carbide (B_4C), effective in neutron shielding, were considered as coating materials for disposal containers. Structural analysis was conducted using the commercial software ANSYS, and shielding analysis was performed using MicroShield. In this analysis, a 200 L disposal container was modeled, and a coating thickness of 0.35 mm was applied.

The aimed of this study was to improve the structural shielding performance of disposal containers through the application of coatings. WC and B_4C were used as coating materials, and it was confirmed that the application of coatings resulted in improved structural and shielding performance. With improved shielding performance, it is possible to increase the volume of the internal container by adjusting the thickness of the shielding material. This can result in an increase in the amount of waste that can be loaded into the disposal container.

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Keywords: Sealed radioactive source, Waste treatment, RI waste, Disposal container

Review of Overseas Radwaste Disposal Site WAC

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Currently, the domestic low and intermediate level radioactive waste disposal site was completed in 2015 and is in operation. The domestic radwaste disposal site establish waste acceptance criteria (WAC) for various radwaste that are currently in operation or generated during decommissioning from radwaste generators and will take over the radwaste. Until now, radwaste disposal site have mainly received dry active waste, so this study was conducted to confirm overseas cases for future disposal of various wastes.

The El Cabril disposal site in Spain has been in operation since 1992. In the case of El Cabril's WAC, a preliminary safety analysis report and a draft of the WAC were developed through basic design engineering in 1987-1988. Based on this, negotiations began with radwaste generators on WAC and waste acquisition methodology. In 1992, it first obtained an operating license through official approval from safety authorities. Initial basis for El Cabril WAC was developed based on safety assessment of the disposal site and the French FRS II.2.e. Additionally, legacy waste, technology that can be processed during nuclear power plant operation and 200 liter waste container were considered.

In the future, it will be used to develop treatment and disposal technology for domestic radwaste through research on WAC and operation case of additional overseas disposal sites.

Keywords: Radioactive waste, Disposal site, WAC

Case Study on the Final Disposal of Radioactive Concrete Waste at KAERI

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Various types of radioactive waste are temporarily stored at the Korea Atomic Energy Research Institute (KAERI) and among them, a large amount of concrete waste is included. Concrete waste was a waste type that had never been delivered to a final disposal site from the Daejeon headquarters of the KAERI before 2023. So, it was necessary to reduce the amount of radioactive concrete waste through the final disposal. In 2023, KAERI carried out the work for delivering the radioactive concrete waste to the final disposal site. This paper briefly reviews the concrete waste treatment procedure for the final disposal.

The concrete waste subject to disposal was the waste generated during the environmental maintenance work around the research reactor in Seoul. Because the waste had been stored for a long time since it was first collected in a 200 L drum, it was repackaged to meet the waste acceptance criteria (WAC) of the disposal site. For repackaging, the waste packages were opened and poured into a work tray and distributed evenly. And then the characterization and repackaging were proceeded as follows. First, samples for radionuclide analysis were collected from at least four locations, focusing on highly contaminated areas among the waste, and free water test samples were collected. The contents of the waste were visually inspected to confirm that it did not contain foreign substances or hazardous substances, and if such substances were included, they were removed. Afterwards, the particulate matter was pre-sorted by a net sieve ($1\text{ cm} \times 1\text{ cm}$) to ensure that no particulate matter was included according to WAC. The waste classified by particle size was collected in a 200 L drum for final disposal, and samples for sieving test were collected during the waste collection. The waste collection process was recorded through step-by-step packaging photography, and after packaging was completed, characteristics such as drum height and weight were measured. Characterization was completed by conducting additional tests such as evaluating the free water content and confirming the presence of particulate matter using test samples. Among the concrete wastes packaged through the above procedures, 50 waste packages were successfully delivered to the disposal site in 2023. The KAERI plans to continuously package and dispose of the remaining concrete waste and also plans to continue to permanently dispose of other types of waste (soil, metal, etc.). Through this, it is expected that the amount of radioactive waste stored in KAERI will be reduced gradually.

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Keywords: Radioactive concrete waste, Final disposal, Waste characterization, Waste acceptance criteria

A Conceptual Design for Disposal of PE-HIC in Disposal Facility

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During the operation of nuclear power plant (NPP), lots of radioactive wastes are generated: dry active waste (DAW), spent resin, boron concentrates, etc. Various treatment techniques; such as cement solidification, polymer solidification, paraffin solidification, packaging in high integrity container (HIC), etc.; have been applied to efficiently manage the spent resin. Currently, packaging in polyethylene (PE) HIC is a primary treatment technique in Korea. The spent resin is transferred to PE-HIC from high activity spent resin storage tank(s). The spent resin drying system (SRDS) is applied to dewater and dry of free-standing water to satisfy the waste acceptance criteria (WAC) of Korea. The generated PE-HICs have been stored in radioactive waste storage building after generation. In many disposal facilities in the United States, including barnwell disposal facility, clive disposal facility, and WCS Andrews, PE-HICs are accepted and disposed. In order to offer substantial structural integrity, overpack structure is applied in many cases.

In this study, conceptual design of overpack for PE-HIC disposal and long-term integrity of PE will be discussed.

Keywords: PE-HIC, Spent resin, SRDS, Disposal

Preliminary Analysis of Characterization Terms to Derive a Disposal Concept (Draft) for Long-Term Storage RI Sealed Source

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As there is no final disposal plan for the disused sealed sources in Korea, they have been accepted by the Korea Radioactive waste Agency (KORAD) and are being stored for a long time in the acceptance and inspection building at Low and Intermediate Level Radioactive Waste Disposal Facility in Gyeong-ju. Accordingly, it is necessary to derive a disposal concept and establish a disposal plan (draft) for disused sealed sources in order to solve the radiation exposure of workers and the reduction of storage space due to long-term storage. Existing disused sealed source characterization has been analyzed physical characteristics in terms of storage/management (package size, weight, shape) and radioactive characteristics (nuclide, half-life, radioactive amount, specific activity) based on disused sealed source information provided by the generator. However, in order to derive a disposal concept (draft) considering final disposal, the physical characteristics of the sealed source (size and properties of the source (solid/gas), the characteristics of the sealing material (size, shape, material) and radiological characteristics (amount of radioactivity at the time of disposal), nuclide inventory by unit disposal container in terms of satisfying the acceptance criteria, etc.) and Category 1 to 5 classification required by the IAEA are required.

As a result of physical characteristic analysis, the most frequently occurring ^{60}Co and ^{63}Ni were analyzed in approximately 1,300 drums. As a result of radiological characteristic analysis, ^{125}I and ^{129}I that are currently stored with a half-life of less than 100 days have reached the clearance level and are excluded from final disposal. ^{55}Fe and ^{60}Co with a half-life of 5 years or less and ^3H , ^{85}Kr , ^{90}Sr , and ^{133}Ba with a half-life of 10 to 30 years can be classified as objects of disposal. It is analyzed that the disposal amount of nuclides with little effect on radioactive decay will be limited during the repository operation period and institutional management period. The radioactivity of the source in the disposal container that satisfies the concentration limit of the 1st phase silo disposal facility and the surface dose of 2 mSv/h of the package is analyzed as a maximum of 1 mCi for ^{60}Co (representative gamma ray source). The preliminary analysis results of this study will be used for the design of a conceptual container for final disposal, the establishment of a storage/disposal scenario, and preliminary safety evaluation of disposal facility (safety criteria: normal scenario 1 mSv/y, abnormal scenario 0.1 mSv/y).

Acknowledgements

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Keywords: Disused sealed source, Physical/radioactive characteristics, Final disposal concept, Disposal container, Storage/disposal scenario

Proposal of Design Requirements for the Development of Intermediate Level Waste Containers

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The decommissioning of nuclear power plants is expected to produce Intermediate Level Waste (ILW), encompassing materials such as pressure vessels, internal structures, spent resins, and spent filters. Despite of this anticipation, the radiological characteristics of ILW have yet to be fully identified, and comprehensive systems for safe transportation, storage, and disposal are currently absent. In this study, overseas cases of developed ILW containers were reviewed. Through this review, design requirements for the development of ILW containers necessary for safe decommissioning and waste management in Korea were derived.

In the U.K, first generation containers were made of concrete boxes with steel reinforcement, but there were issues with radionuclide migration and corrosion. Second generation containers addressed these issues, featuring modular Box Waste Packages made of 6 mm thick stainless steel supplemented with concrete liners. In the latest generation of ILW containers, a shift towards more robust shielded designs has been observed. These containers are crafted from ductile cast iron without encapsulation, ensuring effective radiation shielding.

In Sweden, ILW is classified according to its short- or long-lived nature, with plans in place to dispose of long-lived ILW to SFL. Five types of containers have been proposed to transport and handling of the long-lived ILW, including those for standard moulds, 200 L drums, 280 L protection drums, shielded containers, and long-term durable containers.

Design requirements for these containers include compliance with national standards, appropriate dimensions, and the use of carbon steel with suitable surface treatment. The domestic ILW disposal strategy aims to minimize waste generation, establish long-term storage plans, ensure safe treatment, storage, and disposal, facilitate inspection and recovery, and provide alternative disposal methods if needed. Central to this strategy is the development of secure ILW containers suitable for packaging and transportation, considering ILW characteristics.

Proposed design requirements for ILW containers include dimensional envelope, structural requirements, and waste form compatibility. By meeting these stringent requirements, the disposal strategy aims to ensure the safe and effective management of ILW, minimizing environmental risks and safeguarding public health and safety throughout the disposal process.

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Keywords: Intermediate level waste, ILW containers, Design requirements, Decommissioning

A Review of the Scenario Selection Method of the Radioactive Waste Disposal Facility

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To dispose of radioactive waste in a radioactive waste repository, it should be treated and packaged in a container appropriately according to each country's Waste Acceptance Criteria (WAC). After the number of drums corresponding to the disposal facility design is disposed of, the disposal facility is closed. After an institutional control period, the radioactive waste is permanently isolated from the biosphere and stored safely. The safety assessment of disposal facilities involves uncertainty because of considering a time of at least 1,000 to 100,000 years or more. Therefore, a scenario selection should be made by probabilistic assessment to derive reasonably possible scenarios considering each disposal site's environment.

Scenarios applied to South Korea are classified into normal, abnormal, and human intrusion. In particular, the human intrusion scenario is divided into four types: well, drilling, post-drilling, and resident. The types of scenarios and scenario selection methodologies adopted by other radioactive waste disposal facilities were investigated to assess the appropriateness of the scenarios applied domestically. The main differences are identified in drilling and well scenarios. In the case of the drilling scenario in South Korea, complete penetration occurs immediately after the institutional control period. However, in overseas countries, the drilling scenario is excluded considering the recognition of disposal facility due to the differences in drilling pressure during drilling operations, or the drilling period is postponed regarding the complete degradation time of the engineering barrier. In addition, in the case of the well scenario in South Korea, it is assumed that radiation exposure caused by well usage occurs immediately after the end of the institutional control period. However, in other countries, the proper time of using well is evaluated and adopted. Therefore, the scenarios applied in South Korea are conservative, and a scenario evaluation that reflects the characteristics of domestic disposal facilities and sites should be conducted by referring to overseas cases.

Keywords: Radioactive waste repository, Safety assessment, Scenario, Site characteristic

A Study on the Statistical Similarity of Waste Streams in Radioactive Waste Characteristics

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A lot of radioactive waste streams are newly defined by where they originate or what their types are. Following this standard, a large numbers of radioactive waste streams are under the management of KORAD. As is well known, for the safe disposal of radioactive waste, the characteristics of each stream must be figured out and submitted to evaluate a long-term safety of disposal facilities. It is obvious that analyzing the characteristics of each stream will cost massive time and efforts. Therefore, it is needed to reduce streams by classifying streams according to characteristics they have. In this study, it is analyzed that the similarity among radioactive waste streams based on the obtained property data using statistical methods. And an approach how to manage the upcoming radioactive waste streams has been suggested.

Keywords: Waste streams, Characteristics, Similarity, Covariance

Manufacturing and Performance Test Method of Concrete Disposal Container

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KORAD has been using concrete disposal containers and covers since 2015 to maintain low radiation exposure levels for workers at the LILW repository and effectively handle the disposal of radioactive waste. The disposal containers are designed in three categories. 16-pack disposal containers for 200 L drums, 9-pack disposal containers for 320 L drums, and 1-pack disposal containers for HIC wastes. Disposal containers are constructed of concrete for radiation shielding, and their design life is set at 60 years. The cover is made of either concrete or steel plates with shielding effects equivalent to concrete. The walls of the disposal container are designed to maintain safety, considering both the upper load and the load after closure. The unit weight of the disposal container is 2.3 tons/m³. The self-weight of each disposal container is about 7.0 tons for a 16-pack, about 5.8 tons for a 9-pack, and about 2.34 tons for a 1-pack disposal container.

The materials for the disposal container fabrication are tested according to ASTM standards. Concrete materials for the disposal container include cement, aggregate, admixtures, and water. Other materials besides concrete include rebar, structural steel, steel plate covers, and bearing pads.

The performance tests for disposal containers include 9 aspects: compressive strength (kg/cm²), unit weight (ton/m³), air content (%), water-cement ratio (%), slump (cm), alkali-silica reaction, rock classification test, pile-up test, and creep test. All criteria must be satisfied for each item.

This paper provides a brief overview of the methods used to conduct material tests and performance evaluations for disposal container materials used in the LILW Repository.

Keywords: Radioactive waste, Concrete disposal container, LILW repository

Introduction of Pigging System for Ensuring the Safety of the Groundwater Drainage System at the Gyeongju Underground Silos Disposal Facility

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Since the commencement of operations at the Gyeongju Underground Silos Disposal Facility in December 2014, an average of approximately 1,655 tons of groundwater has been discharged monthly, highlighting the significant management of the groundwater drainage system. In February 2015, scale formation was initially observed in the groundwater drainage system pipes, and subsequent inspections consistently revealed increasing scale formation attributed to dissolved iron and suspended solids in the groundwater. The occurrence of pipe scaling has resulted in decreased flow efficiency and increased pump load. Furthermore, the inability to determine the influence of pipe scale on pipeline corrosion mechanisms has ultimately made it challenging to ensure the safety of the groundwater drainage system.

To address these problems, various methods for pipe scale removal were tested, including electromagnetic treatment and chemical cleaning, ultimately confirming Pig cleaning as the most effective solution. Additionally, the introduction of In-Line Inspection (ILI) for pipe integrity management was considered. Pig cleaning is a process that inserting a Pig into the target pipe and driving it through by pressure to physically crush and remove scale deposits inside the pipe. It is widely used in the petrochemical plant and gas industry. ILI employs an Intelligent Pig equipped with various sensors and measurement devices to precisely measure the size and location of all defects such as corrosion and erosion while they pass through the pipe and it enables to improve pipe management including vulnerability assessment, proactive reinforcement, and pipe lifespan evaluation using the collected data.

We have completed the design of a Pigging system to enable periodic Pig cleaning and ILI in the groundwater drainage system of the Gyeongju Underground Silos Disposal Facility, and we aim to introduce this system. We believe that this introduction design will serve as a significant reference for the groundwater drainage system design studies for the Deep Geological Disposal Facility along with operational outcomes that will be obtained in the future.

Keywords: Groundwater drainage system, Pipe scale, Pigging system, Pig cleaning, ILI (in-line inspection), Intelligent pig

5분과

제염해체 (ORAL)



Spatial Dose Evaluation of Decommissioning Kori NPP Unit 1 Bioshield

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The dose of the space nearby the bioshield of Kori NPP Unit 1 was evaluated for the preparation of its dismantling associated with worker's radiological safety.

After the permanent shutdown of Kori Nuclear Power Plant Unit 1 in Korea, the necessity of decommissioning is increasing nowadays. Not only for the core and spent nuclear fuel, but the adjacent structures - such as bioshield - its radiological features must be analyzed to make a worker's manual based on the ALARA principle. Four major gamma isotopes were considered - $^{60}_{27}\text{Co}$, $^{134}_{55}\text{Cs}$, $^{152}_{63}\text{Eu}$, $^{154}_{63}\text{Eu}$ with the half-lives of 5.26, 2.064, 13.537 and 8.593 years respectively - for external dose calculation from bioshield. Beta isotopes were excluded as they have short range less than 1 m in air, which would have a negligible effect on external dose on workers. Assuming two cases of immediate and 10 years-delayed dismantling after the permanent shutdown of Kori NPP Unit 1, where 10 years-delayed dismantling meant the bioshield had undergone 10 years of cooling before the actual dismantling activity started, spatial dose and worker's external dose according to the selected scenario were evaluated utilizing VISIPLAN 3D ALARA TOOL which was a dose assessment program for assisting ALARA analyze pre-job works developed by SCKCEN, Belgium.

In isotope concentration in Bioshield, $^{60}_{27}\text{Co}$ takes 99.9% of total activity among other isotopes before cooling and 98.6% after cooling. Both cases (immediate/after cooling) showed a significant low spatial dose rate in outer part of bioshield of factor of $10^{-7}\sim 10^{-8}$ mSv/hr, but it differed in inner part of the bioshield as 9.6 mSv/hr before cooling, 3.6 mSv/hr after 10 years of cooling, approximately in factor of 2.7. The evaluation result would be expected to provide the basis for the determination of optimized concrete cutting method, shielding and working scenario on the bioshield dismantling, which is subject to the satisfaction of the worker's dose limit of the maximum 50 mSv/year and 100 mSv/5 year.

Keywords: Kori unit 1 power plant, External dose evaluation, Worker dose evaluation, VISIPLAN, ALARA

Fabrication and Characterization of Thin Plastic Scintillator for Beta/Gamma Detection

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In order to reuse nuclear facility sites after decommissioning, safety must be secured by conducting a radiological characteristic evaluation. In this study, a scintillator was developed for in-situ radioactivity measurement, including beta nuclides, as a technology for decommissioning site restoration. Unlike gamma nuclides, measurement of beta emitting nuclides is difficult due to the short range. Therefore, a thin plastic scintillator was manufactured to measure beta as well as gamma, and the absorbance and beta/gamma measurement performance of each film were evaluated.

The thin plastic scintillator was manufactured using commercial scintillation materials (PPO (2, 5-diphenyloxazolyl), POPOP (1, 4-bis [2-(phenyloxazolyl)]-benzene)), and an ion exchange solution (InnoSol-C100) was used to fabricate it into a membrane form. However, due to the opacity of the ion exchange solution, the transparency of the manufactured scintillator was low, so it was remanufactured by changing the base matrix to PMMA. After adding and mixing a commercial scintillation material to 20wt% PMMA solution, a silicon pad mold of 10 cm × 10 cm was manufactured, and then the solution was doped into the mold. The silicon pad doped with the mixed solution was dried at room temperature to fabricate thin plastic scintillators of each thickness. Plastic films were manufactured in 3 thicknesses (420 μm, 520 μm, and 720 μm) to optimize the thickness. The absorbance of a thin plastic scintillator was analyzed using a spectrophotometer, and detection characteristics were evaluated using Sr-90 and Cs-137 point sources.

First of all, it was confirmed that the plastic film ultimately transmits light of about 400 nm or more through the FRET (Foster Resonance Energy Transfer) phenomenon of the primary scintillation material (Em. Max. about 350 nm) and the secondary scintillation material (Em. Max. about 420 nm). However, as the thickness increased, the transmittance decreased. Therefore, additional analysis such as solubility analysis of the primary and secondary scintillation materials in the base matrix and confirmation of microbubbles is necessary. As a result of beta/gamma measurement, it was analyzed that it was possible to distinguish between Cs-137 and Sr-90 nuclides through the end point energy and peak of the spectrum, and the possibility of beta measurement of thin plastic films was confirmed.

Based on the evaluation results for each film, the feasibility of using a thin-film scintillator for beta/gamma measurement was confirmed.

Keywords: Decommissioning, Beta/gamma detection, Thin plastic scintillator, Film-type, Transparency

Comparison of Calculated and Measured Radioactivity Values of Kori Unit 1 Baffle Former Bolt

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Accurate estimation of the amount of radioactivity at a decommissioning nuclear power plant is important for establishing dismantling strategies and for appropriating radiowaste disposal costs. When performing radioactivity calculations, detailed plant-specific operation history including fuel cycle-specific neutron flux data must be fully considered, since nuclear power plant have a long and a complicated operation history which may lead to significant error. Additionally, In order to obtain validation or reliability, the calculated values must be compared with measured values. In this study, we compare the calculated radioactivity values with measured radioactivity values for baffle former bolt of Kori Unit 1.

Regarding to radioactivity calculation of the baffle former bolt, we carried out cycle-specific neutron transport calculations using RAPTOR-M3G code with three dimensional octant core model of Kori Unit 1 and generated the calculated specific activity value (Bq/g) of the baffle former bolt. In addition, a method considering detailed operation history was also applied. And we performed radioactivity measurements with sample specimens taken from the baffle former bolt. Radioactivity measurement process and results are also introduced. Finally, we compared the calculated and measured result of the baffle former bolt. There is good agreement between calculated result and measured result.

If this calculation method is applied to the reactor pressure vessel or reactor internals, the results of this study will help provide more reliable and accurate radioactivity calculation result. And it is expected that these results can encourage economic and safe decommissioning of nuclear power plant.

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Keywords: Decommissioning, Radioactivity, Neutron transport, Baffle former bolt

Research Status of Laser Technology Development for Cutting Activated Structures in Nuclear Power Plants

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Recently, a laser cutting technology for cutting activated structures is being considered in the field of nuclear decommissioning.

Laser cutting technology is one of the thermal cutting methods that uses a high-energy beam. This is a technology that irradiates a metal object with a laser to melt it and then removes the melt with high-pressure gas. It has been consistently used in general industry, but there is not much experience in applying it in the field of nuclear decommissioning. Laser cutting technology has faster cutting speed and less secondary waste (Aerosol) than other thermal cutting technologies. In addition, since it is non-contact, it is easy to control and automate, making it advantageous for remote cutting of complex structures.

Here, we present the status of laser technology currently being developed for cutting PWR and PHWR at the Korea Research Institute of Decommissioning. In the case of PWR, laser cutting technology is being considered for underwater laser cutting for RVI, and for PHWR, pipe in-bore laser technology is being applied to cut calandria tubes and pressure tubes. The final goal of underwater laser cutting is 200 mm thick cutting, and cutting of 100 mm thick specimens is currently being performed. Pipe in-bore laser cutting technology is being designed for localization of laser head.

Through continuous research on lasers, we plan to advance the technology into a specialized technology for cutting activated structures.

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Keywords: Laser cutting, Underwater cutting, Pipe cutting, Nuclear decommissioning

Developing a Deep Learning Model for Predicting Internal Decontamination Levels in Radioactively Contaminated Tanks

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This study was focused on enhancing the performance of deep learning models for radioactive decontamination technology, utilizing image data of radioactive sludge waste processed by a rotating nozzle. The dismantling of nuclear power plants is emerging as an important issue both domestically and internationally. The safe and effective treatment of various shapes and sizes of radioactively contaminated tanks generated during the dismantling process have to be conducted for fast and economical nuclear decommissioning. The tanks used in nuclear power plants can vary greatly in capacity, internal contamination conditions, amount of sludge, and state depending on their use. It is expected that a variety of organic substances and inorganic oxide sludge will accumulate inside the tank of the radioactive waste treatment system. This study aims to develop an artificial intelligence-based model to predict the degree of internal decontamination of tanks that cannot be visually inspected, thereby enhancing the safety and efficiency of dismantling work and minimizing processing costs. The deep learning intelligence model can determine the degree of internal decontamination based on internal contamination conditions, using only transmitted image files. This artificial intelligence uses U-Net, an image deep learning technology developed primarily for image segmentation tasks. The study conducted various learning experiments, including k-fold cross-validation, to improve the performance of artificial intelligence models using sludge data. The accuracy and BCE (Binary Cross-Entropy) loss were gradually improved in each fold, through which the model's generalization ability and robustness were evaluated and optimized. The first fold started with an initial accuracy of 76.0% and improved to 82.4%, the second fold from 30.3% to 87.4%, the third fold from 61.7% to 88.4%, the fourth fold from 59.8% to 89.9%, and the fifth fold from 56.5% to 84.8%. It enables real-time monitoring of the condition inside the radioactive contamination tanks without the need for workers to enter, thereby enhancing safety by preventing the risk of exposure that could occur when workers enter the tanks directly for operations. This approach contributes to minimizing the risks associated with handling radioactive materials and maximizing operational efficiency. Its characteristics allow for quick assessment of the degree of decontamination inside tanks that are irregularly decontaminated by rotating nozzles with the monitoring device than handwork monitoring. Accordingly, this model can provide important information for the improvement of decontamination device designs and the development of new decontamination technologies.

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Keywords: Nuclear decommissioning, Artificial intelligence, Image segmentation, Radioactive waste management, Decontamination monitoring

Analysis of the Binding Characteristics Between Radionuclides and Soil Using Sequential Extraction Method

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Soil pollutants, including radioactive elements, exhibit varying degrees of mobility and bioavailability based on their chemical forms and soil properties. Identifying these forms, a process known as speciation, is critical for assessing the risk of soil pollution from minerals and radioactive waste. This characterization is essential for understanding and predicting the behavior of trace elements and radionuclides. A comprehensive analysis of soil requires knowledge of adsorption processes and factors affecting pollutant distribution between solid and liquid phases. Sequential extraction is a laboratory procedure used to fractionate various chemical forms of elements in soil, providing insight into their speciation, mobility, and bioavailability. The importance of sequential extraction lies in its ability to distinguish between the different forms of pollutants, thereby offering a more detailed understanding of their environmental impact and behavior in conjunction with the adsorption factors. To analyze the characteristics of soil at nuclear facility sites, samples were collected from the vicinity of nearby site in Seoul Research Reactor, and the nearby sites of nuclear power plant in Wolsong and Kori, and subjected to physicochemical characterization along with XRF and XRD analyses. For studying soil adsorption, 3 mM concentrations of Cs, Sr, and Co were adsorbed onto the soil, which was then used to evaluate the binding characteristics of Cs, Sr, and Co with the soil through sequential extraction experiments. An improved method based on Schultz (1998) was utilized for sequential extraction, which was carried out in six stages: F1 (Soluble), F2 (Exchangeable), F3 (Bound to organic matter), F4 (Bound to carbonates), F5 (Bound to oxyhydroxides), and F6 (Residual). In Cs-contaminated soils, a significant presence was found in the F2 (Exchangeable) and F3 (Bound to organic matter) fractions, while in Sr-contaminated soils, the F2 (Exchangeable) fraction was predominantly observed, and in Co-contaminated soils, the F2 (Exchangeable) and F4 (Bound to carbonates) fractions were most prevalent. The soils were also tested for desorption rates using 0.1 M HNO₃ after adsorption of 3 mM Cs, Sr, and Co, revealing that Sr-contaminated soils showed desorption rates over 80% higher than those of Cs and Co-contaminated soils, attributed to the presence of over 80% of F1 (Soluble) and F2 (Exchangeable) fractions in the sequential extraction method. This data is expected to be utilized in determining decontamination strategies for decommissioning site soils in the future.

Acknowledgements

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Keywords: Decommissioning site, Environmental remediation, Establishing strategies, Radionuclides, Sequential extraction, Soil washing

Stabilization of Combustible Waste Carbonization Residues Through Zeolite Precipitation and Sintering Technology

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The amount of radioactive waste generated during decommissioning of nuclear power plants is expected to be significant, with combustible waste alone estimated to range from 10,000 to 40,000 drums per unit. Considering the volume of waste generated even during operation, it is imperative to identify an appropriate disposal method to reduce waste. Since combustible waste primarily consists of organic matter, conventional incineration technology poses risk such as air leakage due to the volatilization of hazardous gases and radioactive nuclides, as well as the potential for cumulative pollution of exhaust facilities. To address these challenges, research and development efforts are underway for alternative technologies that carbonize combustible waste rather than incinerating it outright. This approach aims to minimize emissions of hazardous gases and halogenated residual inorganic materials, as well as to capture radionuclides in solutions.

In this study, zeolite was chosen as the primary inorganic sorbent for effectively capturing residual chlorinated inorganic materials and radionuclides in solution. This selection was made in the context of applying alternative technology for incineration to decontamination paper, one of the most combustible wastes. Existing commercially available zeolites can be enhanced by impregnating them with metal ferrocyanide (MFC), thereby increasing their adsorption capacity for radioactive cesium. The resulting precipitated residue forms a more complex glass composite material (GCM) system. To further stabilize this waste material for final disposal, we employed cold pressing and sintering technology to solidify it into a stabilized glass-ceramic form. Additionally, additives were incorporated to prevent the volatilization of radionuclides during sintering, thereby reducing the sintering temperature and enhancing stability.

Acknowledgements

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Keywords: Zeolite, Metal ferrocyanide, Glass composite material, Sintering, Solidification

An Analysis on Guidance for Management of Hazardous Materials During Decommissioning Nuclear Power Plants

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The decommissioning of a nuclear power plants (NPPs) require removal of residual radioactivity and other hazardous materials to below regulatory limits to allow reuse. All NPPs are required to undergo a systematic evaluation of physical plant systems, structures, and components (SSCs), the immediate site surroundings, properties, and the environment for both radiological and non-radiological contamination. The need to integrate non-radiological management into radiological decommissioning can substantially complicate, delay, and increase the cost of decommissioning if not properly managed. Thus, to ensure that scope and complexity of non-radiological decommissioning elements are properly addressed during decommissioning projects, early evaluation, planning and stakeholder engagement are critical. To include an evaluation of plant SSCs, past and current plant operations and site use to identify non-radiological elements of SSCs requiring management and assessment of the potential for non-radiological releases to the site and surrounding environment. Therefore, in areas where both radiological and non-radiological impacts exist, the complexity of the site, materials of concern or mixed waste management can be significantly greater as the driving regulatory element or sequencing of approvals and actions may not be clear. In general, assessment and development of a management plan for non-radiological materials should begin concurrent with planning for management of radiological materials. Ideally, this should occur no later than three years prior to plant shutdown, when possible. Non-radiological aspects of decommissioning are diverse, complex, and unpredictable, warranting early planning and systematic evaluation well in advance of cessation of plant operation if possible. This study presents draws upon general industry decommissioning experience; identifies gaps in existing decommissioning guidance relative to non-radiological decommissioning elements; and provides options, tools, and guidance to improve planning for, and integration of the management of non-radiological decommissioning elements in the decommissioning process. Non-radiological constituents of concerns (NRCOCs) encountered in decommissioning generally fall into the following two broad categories. First, SSCs that contain NRCOCs that must be managed during, or in advance of, plant dismantlement, demolition, or site restoration, or may have resulted in a release to the environment. Second, Current, or historic, plant operations that use, store or disposed of NRCOCs that maintain a potential to, or have resulted in, releases of NRCOCs to the environment. For this purpose, non-radiological material management framework is required and the presented framework consists of the following elements. First, Resources and planning. Brings forth concepts presented earlier in the document into proposed application to support non-radiological decommissioning planning and execution. Second, Site assessment. Presents a framework for site assessment to support evaluation of SSCs and site environmental media. Third, Risk assessment. Presents considerations for establishing site cleanup criteria based on generic default values, quantitative risk assessment and, if required, combined assessment of risk from NRCOCs. Fourth, Remedial technologies and alternative evaluation. Presents remedial technologies employed in the abatement of NRCOCs and the process of feasibility evaluation in support of remedial technology selection. It is important to note that this conceptual approach is provided as a general guide for consideration when developing a site-specific plan. Early evaluation and planning to assess the nature and scope of non-radiological challenges in advance of shutdown and to facilitate effective integration with radiological decommissioning tasks will surely benefit the plant decommissioning process, schedule and budget.

Keywords: Decommissioning, NRCOC (non-radiological constituents of concern), Site assessment

5분과

제염해체 (POSTER)



Development of an Intelligent IoT Sensor System for Real-Time Radiation Measurement

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Nuclear power plant workers inevitably work in high radiation areas during nuclear power plant decommissioning or planned preventive maintenance and are therefore exposed to the risk of radiation overexposure. Because the risk of such overexposure is high, the International Commission on Radiological Protection (ICRP) proposes a limit of 50 mSv per year for radiation workers, 6 mSv per year for frequent visitors and transport workers, and 1 mSv per year for the general public under the ICRP-60 guidelines. Domestic laws also apply this effective dose limit as a regulatory standard.

In particular, currently, entry and exit of workers is controlled through the radiation management area access control system, and radiation exposure dose is managed by registering data from the dosimeters worn by workers.

In this study, an intelligent IoT platform was developed by combining a complex sensor for real-time gamma ray/neutron/temperature measurement, a data storage server, and a smartphone application for real-time monitoring to manage worker radiation exposure. As a communication system, a WiFi Station was placed within the nuclear power plant operator's management area, and when the complex sensor was activated, it was implemented to automatically connect to WiFi and transmit real-time data.

This platform monitors worker violations based on worker safety rules learned in advance by AI and issues warnings when necessary. In addition, real-time alerts are provided by predicting the cumulative exposure amount for each worker and the time that can continue working at the current radiation level.

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Keywords: Intelligent, Real-time, IoT platform, Gammaray/netron, Radiation exposure, Pre-learned

Evaluation of Internal Exposure of Wolsong Unit 1 Reactor Decommissioning

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When dismantling a nuclear reactor, radionuclide in the structure were scattered into the air through cutting contaminated structures, and the internal exposure that could occur through the worker's breathing was evaluated.

In order to derive the initial concentration of nuclide released into the atmosphere during the cutting operation, the process involving cutting of the radiated structure was classified during the dismantling process. The structure to be cut is cut by a mechanical or thermal method by a diamond saw or laser, and amount of secondary waste generated are different depending on the cutting method.

Keywords: Decommissioning, Contaminated structure, Nuclide, Wolsong unit 1

Comparison of Dose Evaluation Methodology for PHWR Decommissioning

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COG is actively conducting research on the operation and decommissioning of CANDU. In particular, Wolsong Unit 1 is a CANDU that is planned to be decommissioned at the earliest point in the world, so it is also showing a lot of interest in COG. KHNP has conducted a joint study with COG on the Conceptual Calandria Segmentation and is separately conducting a internal research project.

The evaluation methods and results were compared with the main factors such as work location, dose rate by work location, work time by process, and number of workers used in the COG joint study and the KHNP internal R&D assessment.

Keywords: COG, Wolsong unit 1, Dose, Decommissioning, Calandria

Selection Process of Potential Radionuclides for the Decommissioning of Wolsong Unit 1

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In the decommissioning of Wolsong Unit 1, a CANDU reactor, the selection of potential radionuclides (PRs) is critical for environmental safety and site release criteria. This study describes the potential nuclide selection process for site safety assessment of CANDU reactors utilizing heavy water. Initially, potential nuclides were initially selected through a comprehensive review of CANDU-type power plant PRs being decommissioned. Subsequently, radionuclides with short half-lives or minimal radiological effects were excluded to focus on nuclides with significant long-term environmental and safety impacts. Lastly, potential nuclides were added to Wolsong Unit 1 in consideration of operation history and characteristic evaluation. This approach is similar to the process of selecting potential nuclides when decommissioning overseas light water reactors, but there are differences in the detailed nuclides because they reflect the characteristics of the reactor. In particular, it is necessary to confirm that there is a difference in potential nuclides between light water reactors and heavy water reactors, and to establish a strategy for decontamination by nuclide type when restoring the heavy water reactor site.

Keywords: CANDU reactor, Decommissioning wolsong unit 1, Potential radionuclides

Hydrogen Isotope Separation Behavior of Porous Media and Structural Stability Under Beta Irradiation

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Tritium, which is discharged from nuclear power plants during operation and decommissioning processes, has a threat to the environment due to its high mobility in environment and radioactivity. Beyond the treatment of radioactive effluent, hydrogen isotopes including tritium and deuterium serve as critical raw materials in industries such as display technology and fusion energy, thus the enrichment and separation of hydrogen isotopes are crucial. Technologies applied to tritium enrichment and recovery include water distillation, cryogenic distillation, and catalytic exchange processes. However, these methods have limitations like high operational costs, making them impractical for the enrichment and separation of tritium in low concentrations and large volumes. Consequently, adsorbent and separation media aiming for cost-effective separation treatments and efficient mass production have been continuously researched. Materials such as porous silica, amorphous porous carbon, zeolites, metal-organic frameworks (MOFs), polymer resins, and graphene oxide membranes are explored for separation and adsorptive removal of the tritium. Considering that the tritium emits beta radiation, it is noteworthy to investigating the structural changes and hydrogen isotope separation capabilities of porous adsorbents under radiation exposure.

In this work, we introduce variation in selective adsorption of hydrogen isotopes in porous media upon beta irradiation. Porous media, including zeolites, MOFs like HKUST-1, MIL-53 (Al), and ZIF-8, mesoporous silica, and activated carbon, are selected and characterized. The accumulated radiation dose of the porous media is systematically increased up to 18 MGy using an electron accelerator at the Advanced Radiation Technology Institute of Korea Atomic Energy Research Institute. The structural changes are identified by FTIR and PXRD measurements. The impact of beta irradiation on the surface area and porosity of the porous materials is analyzed through nitrogen gas adsorption experiments. The trend of selective hydrogen isotope adsorption capacity in the porous adsorbents following the radiation exposure is revealed through the individual adsorption of hydrogen and deuterium and the study of ideal adsorbed solution theory (IAST).

Acknowledgements

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Keywords: Hydrogen isotope separation, Tritium, Zeolite, Metal-organic framework, Radiation adsorbed dose

Investigation on Generation Characteristics of Radioactive Air Pollutant and Air Purification Technology During Nuclear Power Plant Decommissioning

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Nuclear power plants are important energy sources that produce large amounts of electrical energy. However, aging nuclear power plants become uneconomical and pose threats to the environment and safety, leading to decommissioning. Buildings and metal structures within nuclear power plants are dismantled using various methods such as mechanical, thermal, and electrical cutting techniques. During this process, radioactive contaminants such as VOCs, aerosols, dust, and gases are generated. Radioactive particulates generated during decommissioning include various isotopes such as Co, Cs, Mn, Sr, and Ni. Radioactive materials accumulate in human tissues and cells through inhalation, directly affecting human tissues and cells, causing DNA damage and cell mutations. To prevent potential generation of radioactive materials and ensure worker safety and protection from exposure during nuclear decommissioning, air purification devices are necessary during operations. Therefore, this study aims to investigate the characteristics of radioactive dust, gas, aerosols, and other airborne pollutants, as well as air purification technologies.

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Keywords: NPP (nuclear power plants), Radioactive contaminants, Air purification technology

An Evaluation Method for Radiological Characteristics to Assess Performance of System Decontamination

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Full system decontamination technology is applied during decommissioning of nuclear power plants to reduce workers' exposure doses and volume of radioactive waste. It is essential to collect data from evaluations of the accurate radiological characteristics of main equipment and piping within decontamination pathway.

The evaluation of radiological characteristics is a vital process that determines radionuclides, levels of radioactivity, and contamination distribution. This is achieved through theoretical calculations based on past operational history, analysis of samples, and in-situ measurements.

The purpose of this study is to propose an evaluation method for radiological characteristics to assess performance before and after system decontamination, based on cases of nuclear power plant decommissioning. This includes review of the radioactive inventory attached to systems, devices, and piping, and examining the specific location and extent of system decontamination.

Additionally, by segmenting the main tasks related to workers' exposure during decommissioning process and providing guidelines, we aim to establish an effective work process plan for system decontamination. Lastly, we intend to utilize this for development of shielding to ensure the safety of workers during dismantling work, along with evaluating the expected exposure doses for workers during future system decontamination.

Acknowledgements

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Keywords: Decommissioning, Exposure, Radiological characteristics, System decontamination

Analysis of the Impact of Nb-94 on the Selection of Radionuclides of Concern for Preparing Wolsong Unit 1 Decommissioning

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The permanently suspended Wolsong unit 1 nuclear power plant is on standby for dismantling.

In the dismantling of a nuclear facility, it is crucial for the regulatory bodies to inspect the completion of the decontamination and to verify the deregulation. Therefore, after the decommissioning of the nuclear power plant, the nuclear facilities must be decontaminated and dismantled to satisfy Derived Concentration Guideline Levels (DCGL) for site release.

The nuclear fuel channel assembly of Wolsong Unit 1, a CANDU-type nuclear power plant, consists of pressure tube and a calandria tube. Zircaloy, a component of the pressure tube, generates radionuclides such as Co-60, Nb-94, Mn-54, Fe-55, Ni-63, etc. due to neutron reaction.

In the case of heavy water reactors, the concentration of Nb-94 is relatively higher than that of other radionuclide due to the thermal neutron absorption reaction compared to PWR, and the half-life of this radionuclide is quite long. ($t_{\frac{1}{2}} = 2 \times 10^4$ years)

This condition could have a substantial impact on the determination of radionuclides of concern for decommissioning preparations, to the extent that it could ignore the importance of other radionuclides.

Therefore, in the dismantling scenario considering dismantling the pressure tube expected to have Nb-94 from the reactor, a scenario was developed by considering the cases with and without Nb-94 in determining the radionuclide of concern. And the effect of the presence or absence of Nb-94 on the determination of the radionuclide of concern was evaluated.

Acknowledgements

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Keywords: Decommissioning, Radionuclides of concern, Zircaloy, DCGL

A Study on the Decay Correction Factor for Comparison With Overseas Cases in the Dose Estimation of Decommissioning Workers

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One of the safety issues in the decommissioning Nuclear Power Plants (NPPs) can largely be related to the workers and residents. During the decommissioning of NPPs, the workers safety against the radioactive contamination of the Systems, Structures, and Components (SSCs) will be important, which can be caused by the performance of work related to Decontamination and Dismantlement (D&D) activities. In the stage of preparing decommissioning planning before starting the decommissioning, the licensee's dose estimation is also a regulatory requirement to ensure safety. However, dose estimations for activities can usually be assessed based on works from the same or similar prior experiences, which is based on reasonable grounds. However, in case of Korea, where decommissioning will be performed for the first time, there is no history to base the previous experience. Even though, it is necessary to compare it with that of previous overseas experiences if we want to roughly confirm the results of dose estimation. Therefore, this study will consider decay correction factor to consider for comparison with overseas cases in estimating worker exposure dose.

First of all, we checked the data containing the contents of the licensee's dose estimation of Kori Unit 1, the first decommissioning nuclear power plant in Korea. Based on this, factors to be considered for comparison with overseas cases were also reviewed. There may be number of factors, but among them, it was judged that a standard needed to be prepared for the timing of work with a large dose effect (i.e., a decay correction factor). Dismantling with a large dose impact of workers generally include Reactor Vessel (RV) and Reactor Vessel Internals (RVIs) with high radiation contamination. Currently, a plan is prepared for the timing of these work, and it can be changeable. In case of Kori Unit 1, it was permanently shut down in 2017, and in the initial plan, major works are expected to proceed in 2025, about eight years after the permanent shut down, but due to the delay in the Spent Nuclear Fuel (SNF) management, it was found that the time can be changed to be 2032.

In this study, we applied the decay factor for comparison with the dose estimation of overseas cases and derived the values of this factor in consideration of the change in the decommissioning schedule. In addition, the effects of dose estimation from 2025 to 2032 were assessed by applying the decay correction factor for changes in the current predictable major dismantling activities. We believe that this results can be used as useful information for comparing does estimations by reflecting domestic circumstances.

Keywords: Decommissioning, Dose estimation, Dismantling, Decay correction factor

Review of the Radionuclides of Interest by Media Considered at the Time of License Termination for the Decommissioning Nuclear Power Plants

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It is necessary to confirm the safety in terms of site release of decommissioning nuclear power plants.

Terms that often appear here include dose assessment, site release criteria expressed in dose, and DCGL. Sites and buildings are representative of the last remaining media after nuclear facilities are dismantled. In some cases, there are media that remain underground, such as concrete structures, buried pipes, and groundwater. Studies related to their safety may have several aspects, and the selection of target radionuclides should be considered in common. Therefore, this study intends to review the cases of previous nuclear power plants abroad for radionuclides of interest in media considered in the site release stage.

We investigated four nuclear power plants (Zion, Rancho Seco, Connecticut Yankee, and Humboldt Bay) in the United States, which are relatively easy to identify in the license termination literature. There were differences in the media considered for each site, but there were many similarities. In general, evaluation was performed on the site and building, indicating that some of the target radionuclides were considered accordingly. In addition, it can be seen that the radionuclides considered are different for each media because the radionuclide contaminated is different for each media. We represented the radionuclides of interest considered for each media by site, and at the same time, the radionuclides generally considered for media were also compared by nuclear power plant.

There is no experience in decommissioning commercial nuclear power plants in Korea yet, and the stage of site release could come as a distant future. However, the deregulation of the site should be considered from the stage of preparation for decommissioning and has an impact on the final stage. In addition, it is necessary to sufficiently identify the media that should be considered until the site release during the preparation or process of dismantling. Therefore, it is expected that the information on the radionuclides of interest by the media of overseas nuclear power plants surveyed in this study can be useful as data for the preparation of decommissioning domestic nuclear power plants.

Keywords: Decommissioning, Radionuclide of interest, License termination, Site release media

Fundamental Study on Cutting of Zirconium Alloys With a High-Power Fiber Laser for Decommissioning of Heavy Water Reactors

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Laser cutting is a valuable tool for nuclear decommissioning applications. With a laser, thick metal can be cut rapidly, and due to its thermal cutting nature, there is minimal reaction forces, eliminating the need for a high-load manipulator. Additionally, the narrow kerf width results in reduced secondary waste, and utilizing optical fiber delivery facilitates cutting from a distance with only a small head.

For this reason, numerous researchers have undertaken studies to implement laser cutting in nuclear decommissioning. However, the majority of these studies focused on steels, particularly stainless steel, which constitute the majority of nuclear reactor components. Nonetheless, the pressure tube and calandria tube of the heavy water reactor fuel channel are made of zirconium alloys. Zr-2.5Nb alloy is utilized for pressure tubes, while Zircaloy-2 is used for calandria tube. Therefore, for cutting nuclear fuel channels in heavy water reactors, it is imperative to study laser cutting techniques to these zirconium alloys.

In this work, we performed a cutting study on 10 mm thick zirconium alloys using a high-power fiber laser. The maximum cutting speed was determined for both Zircaloy-2 and Zr-2.5Nb alloy. Furthermore, we evaluated and summarized the level of secondary emissions generated during the process by measuring the kerf width. These results are expected to serve as fundamental data for future applications for decommissioning of heavy water reactors.

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Keywords: Laser cutting, Laser processing, Nuclear decommissioning, Nuclear dismantling

A Study on the Aerosol and Particle Collection Method and Device of Underwater Operations During Laser Cutting of Activated Structures Around the Reactor

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KHNP is planning to underwater cut and dismantle the activated reactor internal structures of the permanently shut down Kori Unit 1 to ensure radiational safety.

The laser Nd:YAG cutting technology used in nuclear facilities decommissioning emits aerosol particles even while performing underwater operations, and these aerosol particles are released by moving to the surface through various bubbles in the water.

Since the released bubbles contain radioactive aerosol particles, they must be collected using an appropriate method to ensure radiological safety.

Since it is difficult to collect these aerosol particles underwater during laser cutting, the characteristics of aerosol particles emitted through bubbles on the surface of the water must be predicted and an appropriate capture method must be applied.

Therefore, the aerosol particle confinement space and air venting systems must be designed to collect radioactive aerosol particles and ensure radiological safety of the laser cutting work spaces.

We have designed aerosol particle collection devices by considering occupational risks of the location and movement of radiation workers during underwater laser cutting of the irradiated internal structure of the Kori Unit 1 nuclear reactor.

To ensure radiological safety through confinement of radionuclides during the decommissioning of nuclear facilities, it is necessary to predict how many radioactive aerosol particles can be generated and collected in HEPAs before lead to its clogging and rupture, and, based on this prediction the ventilation systems has been designed to maintaining appropriate efficiency of HEPAs.

In addition, the cutting position in the water can vary depending on the type of underwater radioactive waste structures, and accordingly, air ventilation systems has been designed with different hood shapes to collect aerosol particles for each type.

The goal of this paper is to present research results on methods for capturing aerosol particles emitted when underwater laser cutting methods are used for the dismantling of radioactive structures in nuclear facilities.

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Keywords: Decommissioning, Laser cutting, Radioactive waste, Aerosol, Particle, Collection

Study of the Dismantling Sequence of Wolseong Unit 1 Reactor Cooling System From a Radiation Safety and Economic Perspective

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With the decision on the immediate dismantling (DECON) strategy for Wolseong Unit 1 in 2019, Wolseong Unit 1 will become the first commercial Pressurized Heavy Water Reactor (PHWR) to be decommissioned worldwide.

Decommissioning a nuclear power plant is very different from decommissioning a typical structure because it involves dismantling a structure that is radiation generating or contaminated by the radioactive sources. Since the cooling system facilities are extensively contaminated with radioactive substances via heavy water and there is the risk of radiation exposure to workers, the decommissioning sequence needs to be reviewed from a radiation safety perspective. In this study, we examine the dismantling sequence of the cooling system facilities of the Wolseong Unit 1.

The main facilities of the cooling system of the PHWR are steam generator (SG), heat transport pump (HTP), pressurizer (PZR), degasser, and D₂O tank. Considering the radiation safety aspect, the dismantling of the equipment with high radiation is expected to reduce the cumulative radiation dose to the workers by dismantling the equipment with high radiation first.

The HTS of PHWR made of carbon steel has an oxide film formed by dissolved iron ion (magnetite, Fe₃O₄) deposits in the piping and equipment. On the outer layer of the magnetite film, the film of oxide in the form of magnetite and solidified metal oxide (NiOFe₂O₃) is formed. The radiation generated in components of power plant comes from the oxide film formed by the precipitate, which is due to the decrease in the solubility of iron ions as the temperature of the heavy water decreases and more precipitates are generated. As a result, the lower temperature parts of the HTS emit more radiation than the higher temperature parts.

From an economic point of view, it is advantageous to group units that can work in parallel using same equipment. In PHWR, auxiliary cranes are installed on top of the main crane when dismantling large components. D₂O tank is the last to be dismantled as it is isolated and cannot be used with the same equipment.

In conclusion, it would be efficient to start dismantling with the HTP, which has the lowest temperature of the heavy water, followed by the degasser, PZR, and then the SG, which has the highest temperature. And for economic reasons, D₂O tank is the last order for dismantling.

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Keywords: Pressurized heavy water reactor decommissioning, Reactor cooling system, Radiation safety

Plasma Processing Technology Development and Operation Status

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Korea Hydro & Nuclear Power Co., Ltd. can reliably dispose of radioactive waste generated from nuclear power plants Plasma treatment technology was developed. Plasma treatment technology uses the electric arc phenomenon to produce radioactive waste such as concrete and insulation materials with heat of 1,600°C or higher. It is a technology that reduces the volume to more than 1/5 by melting. Targeting various types of wastes (concrete, insulation materials: glass fiber, perlite, filter, etc.) generated during operation and dismantling. Plasma treatment demonstration test (about 1,600 hours) is being conducted. To establish a standard treatment process for each waste, torch output, melting furnace internal temperature, and melt discharge characteristics are evaluated after the demonstration test, and to secure stability for long-term continuous operation, the material of core facilities (plasma torch, etc.) is improved. In addition, the optimal design data of the exhaust gas treatment system for each waste is derived through the measurement and analysis of environmentally hazardous substances of exhaust gases.

KHNP will continue to research plasma treatment technology and secure optimal radioactive waste management technology through facility stability verification.

Keywords: Plasma, Waste, Volume, Melting

A Status of the Global Demonstration Facilities of Advanced Decommissioning Equipments

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Overseas, advanced nuclear decommissioning technology is being actively developed using the 4th Industrial Revolution technology such as robots and digital twin. In accordance with the international trend, industry-academic-research institute cooperation in Korea is also promoting the development and demonstration of nuclear decommissioning equipment using ICT technology. The developing technologies and equipments will be used not only to decommissioning domestic nuclear power plants, but also to enter the global decommissioning market in the future.

In particular, it plans to apply advanced technology (Digital twin, etc.) to the decommissioning of Kori unit 1 and Wolsung unit 1 at a time when there is no experience in decommissioning of commercial NPP. Thus, it is very important how well we demonstrate the reliability and safety of technologies and equipments prior to field application. Even in the global decommissioning market, it is difficult for developed equipment to be competitive without the experience in decommissioning field or empirical data.

Therefore, it is necessary to benchmark how the demonstration of decommissioning technologies and equipments applied with advanced technology is carried out overseas. In this study, it had investigated the cases of the establishment and operation of demonstration facilities of overseas on advanced nuclear decommissioning technologies and equipments and drew implications.

Acknowledgements

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Keywords: Decommissioning, Demonstration, ICT technology, Decommissioning equipments

A Case Study on Considerations When Assessing Decontamination Technologies for Decommissioning

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According to the Article 85–11 (Decommissioning Strategy, etc.) of the Regulation on Technical Standards for Nuclear Power Plants, Etc., ‘Proven technology must be applied to decommissioning of nuclear reactor facilities... (the rest omitted)’. However, because there has been no experience in decommissioning commercial nuclear power plants in Republic of Korea, empirical data, whether the technology is suitable to be utilized at the decommissioning site, has not been sufficiently secured. In this study, we investigated into several papers related to the decontamination technology demonstration, in order to secure considerations which could be utilized at decommissioning sites. According to the results, DOE and OECD/NEA commonly conducted that not only decontamination performance but also ‘Ease of operation’, ‘Secondary waste’, ‘Workers’ safety’, ‘Cost’ are comprehensively evaluated during the technology demonstration process. Based on this study, Korea Research Institute of Decommissioning will derive requirements which could be used to demonstrate various technologies utilized at decommissioning projects from domestic&global nuclear power plants.

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Keywords: Decommissioning, Decontamination, Demonstration, Requirements

Case Study of Aerosol Production and Radionuclide Contamination in Decommissioning of NPP

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When decommissioning a nuclear power plant (NPP), dismantling and decontamination are the main processes adopted to dispose of contaminated structures. Of these, the dismantling of structures is done by physical and thermal cutting process. However, during the cutting process these aerosols are very fine in size, $< 10 \mu\text{m}$ as a suspension of particles or droplets in the air and include airborne dusts, mists, fumes, or smoke. Particularly aerosols can be formed from the vaporization and condensation of metal at the cutting edge into ultra-fine particles and are the primary source of increased internal exposure to workers through inhalation. In addition, reactor vessel (RV), steam generators (SG), and piping in the primary system are major sources of metallic radioactive waste. These metallic wastes have a very thin layer of radionuclides on the inner surface of the pipes by the Chalk River Unidentified Deposit (CRUD) generated during plant operation, which is the main source of radioactive aerosols during the cutting process. This surface layer grows to approximately $70\sim 100 \mu\text{m}$ and forms a layer of radioactive corrosion products, which, depending on the nature of the pipe used, can vary in radionuclides such as ^{54}Mn , ^{55}Fe , ^{58}Co , ^{60}Co , ^{63}Ni etc.

Therefore, we present the comprehensive aerosol productions through case study research in the specific conditions of NPP decommissioning. First, the mechanism of aerosol generation is described. The main factors are the size and mass of the generated aerosol according to the cutting material (Carbon Steel, Stainless Steel etc) with cutting method. Second, the possible radionuclides generated according to the material properties of the structure are inferred. The source of radionuclide generation and surface contamination of the structure comprising the primary system are investigated. Therefore, the generation rate of radionuclides according to the cutting material can be estimated during the cutting process, and it can be used to evaluate the internal exposure of the operator to possible radioactive aerosols.

Finally, it is expected that the physicochemical data of the generated aerosol can be used to reduce the decommissioning cost efficiently and safely by controlling the internal exposure of workers during the decommissioning process.

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Keywords: Aerosol, Decommissioning, Radionuclide, Cutting process

Isotope Selective Electrical Transmission of Hydrogen (H/D) in Liquid Electrolyte Through PTFE Reinforced PFSA Polymer-Graphene Composite Membrane

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Electrochemical separation of isotope mixture of gaseous hydrogen (H_2/D_2) has been investigated using perfluoro-sulfonic acid polymer (Nafion) and 2-dimensional crystals. Building upon the Lozada-hidalgo's hypothesis with modification by Xiangrui Zhang, a theoretical selectivity of 31.7 for tritium (H/T) is expected. Therefore, this Nafion | graphene structure has potential to be applied for tritium separation from radioactive wastewater, but more practical modification of membrane is essential, as swelling of Nafion by liquid water induces crack inside the graphene. In this study, we fabricated PTFE reinforced Nafion (PN) by pore-filling method, and successfully transferred monolayer graphene on it by PMMA assisted method (PN|G).

The quality of graphene was characterized by Raman spectroscopy with a 633 nm laser. For Nafion film without any mechanical support, the hydrogen isotope separation behavior was not reproducible and the proton selectivity against other cationic radionuclide was not maintained, because swelling of Nafion induced many cracks on monolayer graphene. On the other hand, the PN, wherein the dimensional expansion of Nafion is limited by the stiffness of the porous substrate, integrity of graphene was preserved. The proton selectivity of developed membranes were experimentally evaluated by cyclic voltammetry (± 0.1 V, 5 mV/s) and electrochemical impedance spectrometry (EIS). Target membrane (PN and PN|G) were positioned between two chambers filled with 0.05 M of H_2SO_4 , D_2SO_4 , and K_2SO_4 electrolytes. The conductivity of H^+ , D^+ , and K^+ through graphene, Nafion, and electrolyte were calculated by serial circuit model, and H^+ selectivity against D^+ and K^+ of each component was determined by conductivity ratio of each ion. The H/K conductivity ratio was found to be 9 for PN and 60 for PN|G, respectively demonstrating high proton selectivity of Nafion and the sub-atomic selectivity of graphene. The H/D hydrogen isotope separation factor was 2 for PN. However, incorporation of graphene into the transport path of hydrogen isotopes increased the separation factor to 6.0 (PN|G). This result well corresponds to the H/D selectivity through free-standing CVD graphene on porous polyimide track etched membrane in liquid water, reported on previous work. The developed graphene-based membrane is expected to be used for new tritium separation process based on the polymer electrolyte membrane water electrolysis (PEMWE) system.

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Keywords: Hydrogen, Deuterium, Tritium, Isotope separation, 2D material, Polymer electrolyte

Investigating Metal Oxide Solubility in Choline Chloride-based Deep Eutectic Solvents for Potential Selective Spent Catalyst Recovery

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In the catalyst industry, there has been little or no attention paid to the recycling of waste catalysts containing hazardous substances, specifically those that can severely impact the environment upon release, often resulting in improper disposal in landfills. Notably, uranium catalysts, widely used in various industries, pose significant challenges for recycling, separation, and purification due to their high chemical stability. A South Korean textile manufacturer generated approximately 7,100 tons/10 years of these catalysts as waste, which is classified as radioactive waste. These spent catalysts require disposal under strict regulations and procedures, yet standards for their proper disposal are lacking due to their complex nature. The SiO_2 support matrix in uranium catalysts, constituting up to 70% of the total volume, presents an opportunity for waste volume reduction and cost savings in radioactive waste management if it can be separated and disposed of as non-hazardous waste.

Deep eutectic solvents (DESs), created by combining a hydrogen bond donor (HBD) with a hydrogen bond acceptor (HBA) and known for their low melting points, eco-friendliness, and non-toxicity, excel at dissolving metal oxides and are well-suited for extracting metal oxides from uranium catalysts. Choline chloride (ChCl), a widely used HBA in DES synthesis, offers advantages such as low cost, mass production capabilities, non-toxicity, and biodegradability. In this study, solubility tests for various metal and metalloid oxides present in uranium catalysts, including Sb_2O_3 and SiO_2 representing the support matrix, using different types of choline chloride based DESs were conducted. Furthermore, the effect of water content on oxide solubility was explored by varying water content (5%, 10%, 20%) in choline chloride:p-toluenesulfonic acid (ChCl:PtsA) mixtures.

The findings revealed that while ChCl:Ethylene glycol and UREA exhibited poor solubility for most oxides, ChCl:PtsA demonstrated satisfactory solubility for the majority of the oxides. This underscores the crucial role of the hydrogen bond donor in dissolving oxides and highlights the need for careful selection of HBDs in DES screening for oxide dissolution. The addition of water to ChCl:PtsA improved solubility for certain oxides including NiO. This study indicates the potential of DESs for recycling spent uranium catalysts and selective metal extraction. However, as solubility tests were conducted on individual metal oxides, future research should investigate solubility in complex matrices containing mixed metal oxides.

Acknowledgements

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Keywords: Decontamination, Deep eutectic solvents, Metal oxides dissolutions, Radioactive wastes, Spent catalyst

Electrochemical Dissolution of Metal Oxides in Deep Eutectic Solvents for Decontamination Applications

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During the decommissioning process of reactors, metal wastes, which constitute about 30% of the total waste, will be generated. Reducing the radioactivity of this metal waste through decontamination can significantly lower the disposal costs by either reducing the total volume of waste destined for disposal or by waste reclassification. As a traditional decontamination method, hydrometallurgy, faces challenges due to the generation of large volumes of acidic waste. As an alternative method to these limitations, solvometallurgy using non-toxic, low cost, and easily synthesized deep eutectic solvents (DES) is being researched. Previous studies have enhanced the solubility of metal oxides in DES under low temperature conditions by up to 10-100 times using an electrochemical method called paint casting. In this study, a three-electrode system was used to observe the changes in metal oxide solubility at various potentials, to determine the optimal conditions for electrochemical decontamination. In the experiment ethaline, a mixture of choline chloride and ethylene glycol at a 1:2 molar ratio, and metal oxides such as copper oxide, nickel oxide and cobalt oxide were used. For the electrochemical setup, IrO-coated Ti mesh electrodes were used as the working electrode (W.E.) and counter electrode (C.E.). The reference electrode (R.E) was an Ag/Ag⁺ electrode created by immersing an Ag wire in ethaline dissolved with 0.1 M AgCl. A paste, made by mixing metal oxide and DES in a 1:1 mass ratio, is painted onto the W.E surface and covered with filter paper to prevent leakage into the electrolyte. Oxidative dissolution of the metal oxides is induced by applying an oxidation potential at 50°C for 24 hours. Each experiment was carried out at least twice to ensure reproducibility. Samples were taken at regular intervals to measure solubility changes over time, analyzed using inductively coupled plasma optical emission spectroscopy (ICP-OES) and ultraviolet-visible spectroscopy (UV-vis). These results will confirm the changes in solubility of metal oxides through electrochemical methods and help select the optimal conditions for decontamination.

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Keywords: Radioactive waste, Decontamination, Deep eutectic solvents, Electrochemistry

Fabrication of a Nanofibrous Pt/Ir-RuO₂ Electrode for Treatment of Organic Decontamination Waste

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A wide variety of chelating agent for decontamination are used in nuclear facilities to remove radionuclides. After decontamination, some of these agents such as ethylenediaminetetraacetic acid (EDTA) and nitrilotriacetic acid (NTA), citric acid, and sodium dodecyl benzene sulfonate (SDBS) are dissolved in water. However, these pollutants can cause several issues during post-processing. Electro-oxidation (EO) is cost-effective treatment to decomposing these organic pollutants. It can also mineralize organic pollutants with a small or no amount of sludge. Therefore, we presented three strategies for fabricating electrodes that enable efficient EO. Nanofibers offered a large specific surface area. RuO₂ and IrO₂ have been shown to be excellent electrocatalysts. The Pt intermediate layer was added with a novel technique to increase structural stability and electrical conductivity.

The nanofibrous template was fabricated by electrospinning of poly (vinyl alcohol) (PVA) solution on a Ti substrate. Pt was deposited onto the PVA nanofibers and then calcined to obtain Pt nanofibers. Then we obtain Pt/Ir-RuO₂ electrode by electrodeposition of the RuO₂ and IrO₂ catalysts onto Pt nanofibers. The study investigated the electrochemical performance of different ratios of Ru and Ir, specifically 10:0, 9:1, and 8:2. The results showed that the ratio of 9:1 had the best performance and was therefore selected. The nanofibers electrode with a Ru:Ir ratio of 9:1 (Ru₉Ir₁-NF) exhibited an electrochemical activation area 1.7 times larger and a charge transfer resistance 4.7 times smaller than the flat-type electrode. Evaluation of the EO performance of Ru₉Ir₁-NF was carried out by degrading 100 ppm of organic pollutants (SDBS, triton X-100, EDTA and NTA) with electrolyte (0.1 M NaCl). More than 80% of SDBS was decomposed within 4 h under the galvanostatic condition of 50 mA/cm². Chemical oxygen demand measurements confirmed that triton X-100, EDTA and NTA decomposed by more than 99% within 24 h, 4 h, and 6 h, respectively, under the galvanostatic condition of 30 mA/cm². Decomposition of radioactive waste was achieved by decomposing more than 99% of NTA combined with cobalt within 4 h in laboratory scale.

Keywords: Electro-oxidation, Iridium-ruthenium oxide, Nanofiber, Organic decontamination waste

Prussian Blue Decoration on the Polyethylenimine Grafted Polyacrylonitrile Nanofiber for the Enhanced Removal of Radioactive Cesium From Water

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In the event of a large-scale nuclear power plant accident such as Chernobyl and Fukushima disaster, a significant amount of hazardous radioactive isotopes (such as ^{137}Cs , etc.) have been released into the natural environment. Given the presence of substantial competing cations (Na^+ , K^+ , Ca^{2+} , Mg^{2+} , etc.) in seawater/groundwater, there is a need for technology capable of selectively removing radioactive nuclides. Nanofibers manufactured by electrospinning are widely used in research on water treatment, not only as membranes for UF (Ultrafiltration) and NF (Nanofiltration) but also as adsorbents in various studies on wastewater treatment. Among various polymers utilized in nanofiber applications, polyacrylonitrile (PAN) has been selected as the primary material due to its excellent heat resistance, hydrophilicity, and chemical resistance, making it widely used in filtration and reverse osmosis membranes. Polyethyleneimine (PEI) was grafted onto PAN nanofibers through hydrothermal synthesis. Ferrocyanides were coated as mediators to introduce Fe^{3+} ions for producing PAN-PEI-prussian blue (PB) NFs. The attachment of PB was also confirmed through SEM analysis. The removal of Cs^+ was fully investigated in terms of Cs^+ adsorption kinetics, isotherms, and selectivity in solutions with various pH levels and salt concentrations.

Keywords: Polyacrylonitrile nanofiber, Polyethyleneimine, Electrospinning, Cs removal, Prussian blue

The Decontamination Process of Radioactive Metal Waste by Using Inorganic Based Chemical Agent

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During decommissioning of nuclear facilities, the decontamination technology is essential process to reduce occupational exposure of radioactive component and contamination of complex equipment. The selection of decontamination is important for the character and amount of waste generated and minimization of the waste. Among the decontamination technologies, inorganic based chemical agents for decontamination is beneficial for high decontamination efficacy and low secondary waste.

In this study, we have performed the practical decontamination application with radioactive metal specimen. We also evaluated the decontamination effect by using the inorganic chemical solution as a decontamination agent. The decontamination is done with HyBRID process and cerium as oxidant in sulphuric acid. We have shown high efficiency for removing the oxide layers of metal specimen within few decontamination cycles. Some metal specimen has been completely cleared after decontamination (Activity lower than 0.1 Bq/g for ^{60}Co). The detailed results will be discussed in the conference.

Keywords: Decontamination, Inorganic acid solution, Secondary waste, Decontamination factor

Removal of Radioactive Cesium From Surfaces Using Partially Hydrolyzed Poly (Vinyl Acetate)-Borax Complex Containing Cs Adsorbent at Low Temperatures

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In the event of a sudden release of radioactive isotopes to the public, such as from a nuclear accident or terrorist attack, wide-area surface decontamination becomes imperative. To remove radioactive cesium, a spray coating comprising a reversible composite based on partially hydrolyzed poly (vinyl acetate) (PVAc 80) and borax has been developed. To use the hydrogel at lower temperatures, a mixed solvent of water and ethanol was used. Upon simultaneous spraying of aqueous solutions of PVAc 80 and borax containing adsorbents onto the surface, a gel-like coating spontaneously is generated. The Cs adsorbent dispersed within the gel-like coating selectively eliminates Cs-137 from Cs-contaminated surfaces. This gel exhibits good viscoelastic properties and sustains its storage modulus (G') and loss modulus (G'') values even subsequent to the addition of the adsorbent. In vertical stability tests, the gels remained stable without flow for up to 3 hours at a low temperature. In radioactive tests, the gel containing Cs-adsorbent showed good Cs-137 removal efficiency from cement, and excellent removal efficiency of 94.66% from stainless steel. This removal efficiency exceeds that of DeconGel, a commercial surface decontamination coating. Furthermore, the applicability of this gel extends beyond room temperatures to encompass lower temperatures as well. Consequently, our spray coatings are suitable for practical wide-area surface decontamination.

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Keywords: Cs removal, Hydrogel, Partially hydrolyzed poly (vinyl acetate), Spray coating, Surface decontamination

The Removal of Metals [Cd(II), Ni(II), Sb(V)] by Using Metal Oxide Biochar Composite

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This study investigated the application of metal oxide biochar prepared from coffee grounds (SCG) and nano iron(III) oxide (FO) for the removal of cations [Cd(II), Ni(II)], and anions [Sb(V)] from aqueous solutions. Material characterization was performed using XRD, BET, and FE-SEM to analyze the properties of the biochar. Batch adsorption experiments evaluated the effects of pH, adsorption rate, and adsorption isotherms. The addition of FO facilitated pore formation during SCG carbonization, leading to a significant increase in the biochar's specific surface area. Adsorption experiments results showed that biochar exhibited dual functionality in removing both positively charged Cd(II) and Ni(II) and negatively charged Sb(V). Based on these results, metal oxide biochar has cost-effective and environmentally friendly advantages and can be used promise as a viable treatment option for both cationic and anionic radionuclides.

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Keywords: Nano iron oxides, Composite, Co-pyrolysis, Sorption

Effect of Catalyst Deposition Methods on the Long-Term Performance of PEM Electrolyzers for Hydrogen Isotope Separation

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Hydrogen isotopes are important in nuclear fusion, medical, electronics and research applications. Tritium in particular is radioactive, making its separation, recovery, and removal critical. Tritium in the wastewater produced by nuclear reactors must be removed to meet discharge standards, and research is needed to develop efficient separation technology. Among the techniques for separating hydrogen isotopes, water electrolysis has a high separation factor and can efficiently enrich tritium or deuterium in water.

In this study, four PEM single cells were fabricated by varying catalyst layer formation methods of membrane electrode assemblies (MEAs), including catalyst coated substrate (CCS) and catalyst coated membrane (CCM), to produce optimal MEAs for hydrogen isotope separation. Electrochemical analysis of the cell was performed with linear sweep voltammetry (LSV), electrochemical impedance spectroscopy (EIS), and cyclic voltammetry (CV). H/D separation factors were analyzed by measuring deuterium content in the gas generated by water electrolysis and the feed water.

After long-term operation of the cell for 100 hours, the electrochemical performance and separation factors were analyzed again. Electrochemical analysis showed that the stability of the cells prepared by the CCS method was higher than that of the CCM method. The active area of anodic catalyst prepared by CCS method was significantly increased after the long-term operation, compared to the cells prepared by the CCM method. The separation factors were successfully maintained even after the long-term operation.

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Keywords: Hydrogen isotope, Stability, Electrode fabrication, Catalyst layer, Water electrolysis

Copper Hexacyanoferrate as an Efficient Capacitive Deionization Electrode for the Removal of Cobalt Ions From Wastewater

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The Co radioisotope (^{60}Co) is frequent in radiotherapy medicines and radioactive wastewater from nuclear power plants. The radioactive half-life of ^{60}Co is 5.271 years, making it difficult to handle and posing a serious threat to health and the environment. Therefore, the removal of radionuclide is extremely necessary before discharging the radioactive effluent. Capacitive deionization (CDI) has recently attracted attention as an effective electrochemical technology for removing ions from wastewater. Especially, asymmetric CDI cells with intercalation-based electrodes can capture many positive ions, providing high specific capacity and enhanced ion removal capacity compared to conventional carbon electrode-based symmetric CDI cells. Prussian blue analogues (PBAs), a representative electrode material, allow cations to be intercalated into the hydrated form without phase transition during capture, leading to fast ion diffusion.

In this study, we investigated the possibility of using copper hexacyanoferrate (CuHCF) to capture Co^{2+} ions from wastewater. CuHCF is synthesized via simple co-precipitation and applied as a novel cathode material in CDI system. X-ray diffraction (XRD) and X-ray photo-electron spectroscopy (XPS) were supported to elucidate the structural evolution of CuHCF during Co^{2+} ion redox process. The well-defined CuHCF with an open structure and two active redox couples exhibited an excellent capacity of 194.1 F g^{-1} at 0.5 A^{-1} . The Co^{2+} ions adsorption amount in the CDI system achieved 141.6 mg g^{-1} under 20 mg L^{-1} aqueous solution, which corresponds to 55% charge efficiency (Λ). This work suggests that CuHCF can be extended as an electrode material for efficient removal of cobalt ions from wastewater.

Keywords: Cobalt ion removal, Electrochemical, Capacitive deionization, Copper hexacyanoferrate (CuHCF)

The Technology Evaluation Factors and Special Considerations for the Environmental Remediation of Radioactively Contaminated Sites

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Major steps in remediation of radioactively contaminated sites are planning for remediation, site characterization, remediation criteria, remediation strategy, implementing remediation actions, and conducting post-remediation activities. Matters which should be considered first at the very beginning of the planning stage. Site characterization is needed to provide sufficient data to take early strategic decisions on the likely environmental remediation activities. If the remediation is justified and any cleanup action optimized, criteria are needed to target remediation activities, to assess performance as the work proceeds, and to verify that the remediation has been achieved at its conclusion. During or after the preliminary site characterization, an engineering study should be conducted to develop remediation options which address the specific contaminant problem and are aimed to reduce radiological and chemical exposure. Options will include engineering approaches and associated technologies. A preliminary selection of options may be made based on several factors including future land use, technical and institutional considerations, public acceptability, cost, regulatory requirements, etc. The implementation of remediation actions includes: procurement of the selected technology; preparation of the site; development of a health and safety plan; development of operations procedures; staff selection and training; completion of site cleanup; verification; waste disposal; and release of the site for any future use. Once remediation activities have been completed and verified, the remediated site can be released for restricted or unrestricted use. However, in most cases it is necessary to impose certain post-remediation activities on the area of concern. These activities will vary in comprehensiveness and duration according to the degree of remediation that has been achieved.

The selection of preferred technologies to solve or mitigate an environmental contamination problem requires the consideration of several factors. The selection process for available technologies can follow various routes. In order to make defensible decisions regarding the selection of the preferred technology, the technology evaluation factors should be treated in an integrated manner. Their consideration in isolation from each other will not address mutually conflicting requirements. The selection of the appropriate technology for remediation requires considerations of several factors such as the ability of the technology to reduce risk, the reliability and maintenance requirements for the technology, the associated cost, the infrastructure available to support the technology, the ease of accessing the technology and associated services, the risk to workers and public safety, the environmental impacts, regulatory acceptance, and community acceptance.

The general approach to remediation of radioactively contaminated sites may require special adaptation to address sites covering very large surface areas, or those which are deep and difficult to access. Small localized sites may benefit from removal or isolation approaches which are not feasible for very large sites. In addition, rigorous quality assurance techniques may be very important to demonstrating success of these projects as remediation criteria approach environmental background values.

Keywords: Remediation, Radioactively contaminated sites

Comparison and Analysis of DCGLs for a Single NPP Site and Multiple NPPs Site Using RESRAD-ONSITE

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In Korea, Kori Unit 1 and Wolsung Unit 1 have been permanently shut down, with plans underway for the decommissioning of these reactors. Decommissioning refers to administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility. As a last part of the decommissioning process, the site and remaining buildings are assessed to determine if they meet release criteria. Based on this assessment, a decision is made regarding regulatory removal.

Derived Concentration Guideline Level (DCGL) serves as a criterion for such release, representing the radioactivity concentration limit for each nuclide, derived from dose criteria for site release. Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), the U.S. detailed guidance on how to demonstrate that a site complies with a radiation dose- or risk-based regulation, recommends deriving DCGL to establish a safety assessment method to ease site regulations. Similarly, in Korea, 'Nuclear Safety and Security Commission Notice No. 2021-11' also applies these principles *mutatis mutandis*.

DCGL must be calculated separately depending on the site subject to release and varies depending on the characteristics of each site and the lifestyle of the person occupied there. Currently, the number of units in each domestic nuclear power plant ranges from a minimum of 2 to a maximum of 8, with multiple units coexisting on the same site. It remains undecided whether the site release process will be conducted for individual units or multiple units during future decommissioning, potentially leading to differences in derived DCGLs based on how the evaluation site is selected during the actual site release permit process.

Accordingly, this study was conducted using RESRAD-offsite to assess the impact of changes in DCGLs of some nuclides due to increases in the area of the site subject to evaluation and changes in site shape. The study aims to assist in the development of site release plans by analyzing the effects of site area and shape.

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Keywords: DCGL, Site release, Area characteristic, RESRAD-offsite

Analysis of Media Subject to DCGL Derivation for Overseas Decommissioning Nuclear Power Plants

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Kori unit 1 and Wolsong unit 1 permanent shutdown in 2017, 2019, respectively. In Korea, decommissioning of nuclear power plant (NPP) can be completed if the results of Final Status Survey (FSS) are less than Derived Concentration Guideline Level (DCGL). Depending on the site reuse scenario, all facilities may be dismantled or some facilities may be remained. In case of overseas NPPs that were completed decommissioning, DCGL were derived for various media. Kori unit 1 and Wolsong unit 1 are planned as brown field, but there is a possibility of green fields in the future. Therefore, in this study, we analyzed overseas decommissioning NPPs that derived DCGL for various media. A number of NPPs have been completed decommissioning worldwide. We analyzed the cases of Yankee Rowe, Connecticut Yankee, and Rancho Seco NPPs that have completed decommissioning in United States, which has the most decommissioning cases. Yankee Rowe and Connecticut Yankee were released green field. Rancho Seco was released brown field and is being used as a power generation facility. Yankee Rowe derived DCGL for Soil, Subsurface partial structure, Concrete debris, Groundwater, and Building surface. Connecticut Yankee derived DCGL for Soil, Groundwater, Building surface, Concrete debris, Basement fill, Buried piping. Rancho Seco derived DCGL for Soil, Building surface, Bulk material. Bulk material is concrete that has potential for volumetric contamination, such as activated concrete. In this study, we analyzed overseas decommissioning NPPs that derived DCGL for various media. In general, DCGL is commonly derived for Soil, Groundwater, Building surface. And DCGL is derived for various media such as Subsurface partial structure, Concrete debris, and Buried piping depending on the situation of the NPP. The results of this study are expected to be used as basic data for deriving DCGL.

Keywords: Decommissioning, DCGL, Media, Kori unit 1, Wolsong unit 1

A Study on Developing Elements of Radiation Scanning Robotics Aligning With the MARSSIM Revision

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The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (NUREG-1575, Revision 1) has not been revised in over 20 years, first published in 1997 and updated in 2001. Although the technical capabilities for radiation measurements and the number of methods for these measurements have increased since MARSSIM was first published according to the technical standards of the 1990s, the manual does not include content for the use of current technology. To address the manual's ambiguity and technological advances, member agencies of the working group, including the U.S. Environmental Protection Agency (EPA), the Nuclear Regulatory Commission (NRC), the Department of Energy (DOE), and the Department of Defense (DoD), have worked together over the past decade to revise MARSSIM and draft was issued on May 2020.

The revisions to MARSSIM are intended to increase clarity and improve flexibility in providing a consistent approach for radiation survey and site investigation of real property (surface soils and building surfaces) to provide defensible and rigorous surveys, especially Final Status Surveys (FSS). The revisions include the improved and novel technologies such as mobile robotic systems, and the guidelines for utilizing robotics proposed. This paper proposes the development elements for scanning robots for the nuclear power plant. Firstly, technologies to ensure postional accuracy, such as GPS, LiDAR (Light Detection And Ranging), local microwave positioning devices and laser sensors, must be employed to obtain the positional data. Secondly, radadiation scanning robot systems to ensure the accuracy compared the sampling are necessary to complement scanning data. Finally, automated data processing and geographic mapping are also developed.

This paper presents the development direction that should be focused on research in the development of nuclear power plant scanning robots for use in nuclear power plant radiation survey and site investigation for operation and decommissioning work.

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Keywords: MARSSIM, Nuclear robotics, Postional accuracy components, Scanning techniques

A Preliminary Study on Considerations of Measurement Quality Objectives for Final Status Survey Based on MARSSIM Revision

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In Korea, the first decommissioning activities of a commercial reactor has begun officially with the goal of safe and successful decommissioning after permanent shutdown of Kori Unit 1. For this, it is one of the most important procedures that demonstration of compliance of dose criteria of which final status of decommissioned site and building based on survey. The Korean government (Nuclear Safety and Security Commission, NSSC) actually stipulate not only the methodology of demonstration of compliance, but also the dose criteria for release and reuse of building and site of nuclear power utilization facilities as 0.25 mSv/y in Notice of NSSC.

On the other hand, one of the key to demonstrate compliance of dose criteria, the data quality objectives (DQO) process which is a series of logical steps to create a plan to ensure that the survey results are of sufficient to support final decision. It is of note that MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) workgroup revised the NUREG-1575 with the objective to include of measurement quality objectives with measurement uncertainty. In particular, the MARSSIM workgroup emphasized the importance of measurement quality objectives (MQO) by officially referring to both DQO and MQO in the revision of MARSSIM. The MARSSIM workgroup explained the changes to update complying with current National Institute of Standards and Technology (NIST) and International Organization for Standardization (ISO) guidance and include most current scientific standards.

Since 2001, the first revision of MARSSIM, continuous research has been conducted in the field of calibration and measurement uncertainty of which focused on verifying and validating the reliability of measurement results. ISO published ISO Standards (i.e. ISO 23547:2022, ISO 8690:2020, ISO 19581:2017, etc.) which covered measurement of radioactivity include surface contamination and rapid screening of material and site. Also, in December 2016, NIST reported their quality manual for measurement services (i.e. NIST-QM-I) focused on their policy and objectives of measurement quality.

Referred to other objectives of MARSSIM revision, the movement of foreign institutes for standardization and regulatory body is expected to continuously apply improved and novel survey technology to the field of nuclear power plant (NPP) decommissioning and consider MQO. From this perspective, this study aims to analyze the major changes for radiation survey and site investigation (RSSI) process with DQO and MQO of MARSSIM revision 2 and derive the considerations and issues for decommissioning of domestic NPPs.

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Keywords: Decommissioning, Site release, Data quality objectives, RSSI, MARSSIM

Predicting the Distribution Coefficients of Strontium and Cobalt Using Hybrid Method With Machine Learning and Experimental Data

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Nuclear facilities face a critical challenge in managing the migration and retention of radioactive contaminants, including strontium (Sr) and cobalt (Co), under various environmental conditions in the event of unforeseen incidents. The distribution coefficient (K_d) plays a pivotal role in understanding the mobility of these contaminants, which is affected by environmental factors. This research aims to predict K_d values for radionuclides in different soils by employing machine learning models. These models were developed using experimental data and publicly available data from the Japan Atomic Energy Agency (JAEA). Specifically, three machine learning models, convolutional neural network (CNN), artificial neural network (ANN), and random forest (RF), were utilized to predict the distribution coefficient (K_d). The training of these models was based on fourteen input variables from the database and experimental findings, including initial contaminant concentration, characteristics of the solid phase, and the conditions of the solution. Preprocessing techniques such as normalization and logarithmic transformation were applied to these variables to improve the models' accuracy. The effectiveness of these models was assessed using the coefficient of determination, revealing that factors such as the type of environmental medium, initial concentration of radionuclides, properties of the solid phase, and conditions of the solution significantly influence the prediction of K_d values. The results of this experiment evaluated the machine learning predictions when experimental data was incorporated alongside the JAEA-SDB, compared to using the JAEA-SDB data alone. The experimental data in addition to the JAEA-SDB data demonstrated an improvement in prediction performance. This improvement suggests that considering the environmental impact on actual sites provides more enhanced data outcomes.

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Keywords: Radionuclides, Soils, Distribution coefficient, Prediction, Machine learning

Evaluation of DCGL for Clearance of Spent Activated Carbon Using RESRAD-Onsite Code

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Among the waste generated by nuclear power plants, impregnated Spent activated carbon used in the air purification system which deteriorates as the number of years of use increases, and is generated dozens of drums every year. Most of the generated Spent activated carbons are Clearance after management in temporary storage in nuclear power plants, but in the case of some Spent activated carbons, the concentration of ^{14}C exceeds the Clearance limit, so they are stored without being able to perform their own disposition. In order to determine compliance with Clearance criteria, either Acceptance concentration or Acceptance dose limits must be met. To assess compliance with Acceptance dose limits, Derived Concentration Guideline Levels (DCGL) evaluation must be conducted. The DCGL defined as the radiological environmental impact assessment must be derived from the evaluation of the exposure dose for the decommissioning site using Residual Radioactivity (RESRAD) code. The Select between the Residential Farmer Scenario and the Industrial Worker Scenario during evaluation significantly influences the resulting outcomes. In brown field (industrial worker) scenarios, the exposed individual usually spends less time on site, and fewer exposure pathways usually are involved than for the resident farmer scenario. Based on the industrial worker scenario, the RESRAD pathways typically suppressed are ingestion pathways (Meat, Plant, Milk) and aquatic foods. In this paper, sensitivity analysis of key exposure pathways and variables was evaluated during the DCGL assessment using RESRAD Onsite. For the creation of input variables, EPRI technical reports and domestic and international research reports were utilized.

As a result, it was confirmed that the DCGL of resident farmer scenario was more conservative compared to the DCGL of industrial worker scenario. In the residential farmer scenario, ingestion of fish was assessed as the primary exposure pathway, with density and thickness of the zone evaluated as sensitive parameters. In the industrial worker scenario, inhalation was assessed as the primary exposure pathway, with density of zone and inhalation rate evaluated as the sensitive parameters. It is expected to be available for the feasibility of verifying cost for release of Spent Activated Carbon evaluation through various scenario evaluations using RESRAD.

Keywords: RESRAD, Spent activated carbon, DCGL, Release, Disposal

Effects of Dose Evaluation by Exposure Pathways According to the Site Reuse Scenario Used in Decommissioning Nuclear Power Plants

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The Derived Concentration Guideline Level (DCGL) is generally calculated using the RESRAD code in several nuclear power plants for reuse of the site and remaining buildings of dismantled nuclear power plants. Each country presents the criteria for the release of the sites as doses, and based on this, DCGL is derived through dose evaluation considering each site's scenario, input variables, and exposure path. The dose evaluation considering these will be evaluated by licensee reflecting the characteristics of the site. At this point, the dose result varies depending on what the site reuse scenario is selected, and the effect on the dose is expressed accordingly. Therefore, this study aims to analyze the change in the input parameter value and the evaluation results for the site reuse scenario considered when evaluating the site reuse of nuclear facilities.

Two main scenarios can be considered for the reuse of the site of decommissioning nuclear power plants: the resident farmer scenario, which represents the most conservative evaluation, and the industrial worker scenario, whose exposure path has been partially removed. The expected exposure pathways from contaminated soil may be external exposure, which is largely occupational exposure from contamination of the soil, and internal exposure consisting of inhalation and ingestion. In addition, the radioactive radionuclide inherent in the soil spreads to the groundwater through the unsaturated and saturated zones under the soil through loss and leaching. In the case of loss, it may be re-suspended into the air or may disappear over a certain amount of time. Therefore, it is a resident farmer scenario that considers all of these effects, and it is evaluated that exposure is caused by the presence of residents in contaminated soil and the consumption of vegetables, livestock, and meat from contaminated soil. In the case of most of the preceding overseas decommissioning nuclear power plants, most residential farmer scenarios representing conservative evaluations were applied to evaluate the long-term effects.

In this study, we evaluated the doses generated during the reuse of the decommissioning nuclear power plant by dividing them by application scenario. In this result, radionuclides that greatly affect the external exposure dose and some that greatly affect internal exposure were classified by radionuclide, and industrial worker scenario had a different effect on the total dose as some exposure pathways were removed in the resident farmer scenario. In order to reflect the legal requirement in the application of the scenario, it is necessary to distinguish whether it is a restricted or unrestricted target for that scenario. In general, resident farmer scenarios will be unrestricted and industrial workers will be restricted. In addition, in the case of restricted conditions according to the nuclear safety and security commission, the application of the exposure dose to the general public should be considered.

Keywords: Decommissioning, DCGL, Site reuse scenarios, Resident farmer, Industrial worker

Analysis of Site Conceptual Model of Basement for Overseas Decommissioning Nuclear Power Plants

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Kori unit 1 and Wolsong unit 1 permanent shutdown in 2017, 2019, respectively. In Korea, decommissioning of nuclear power plant (NPP) can be completed if the results of Final Status Survey (FSS) are less than Derived Concentration Guideline Level (DCGL). Depending on the site reuse scenario, all facilities may be dismantled or some facilities may be remained. Kori unit 1 and Wolsong unit 1 are planned as brown field, and there is a possibility that some structures, pipes, etc. may remain in the site after decommissioning. In the case of overseas decommissioning NPPs, site concept models of basement were established for structures and pipes that could remain underground depending on the decommissioning situation. Therefore, in this study, we analyzed site conceptual model of basement for overseas decommissioning NPPs. Zion, Connecticut Yankee, and Rancho Seco NPPs were analyzed.

Zion NPP assumed that structures and pipes would remain underground after decommissioning. Piping categories were divided into three categories: Buried Piping, Embedded Piping, Penetration piping. Buried piping is below ground pipe located outside of structures and basements and buried in soil. Embedded piping is defined as piping that is located, or embedded, in the concrete basement floors. Penetrations are the portions of various systems piping remaining within building walls, floors, and ceilings after all systems and piping are removed from building interiors. When establishing the site conceptual model of basement, it is assumed that the inner surface contamination, residual radioactivity is distributed uniformly, underground facilities are located below the surface, interior of underground facilities is filled with non-contaminated fill material, non-contaminated covering layer is composed of 3.6 m on the upper part of the underground structure. Connecticut Yankee NPP assumed that structures and pipes would remain underground after decommissioning. When establishing the site conceptual model of basement, it is assumed that the inner surface contamination, concrete volume contamination, residual radioactivity is distributed uniformly, interior of the underground facilities is filled with non-contaminated fill material, non-contaminated covering layer is composed of about 0.7 m on the upper part of the underground structure. Rancho Seco NPP assumed that pipes would remain underground after decommissioning. When establishing the site conceptual model of basement, it is assumed that the inner surface contamination, residual radioactivity is distributed uniformly, underground facilities are located at a depth of approximately 0.3 to 1.0 m below the surface, the interior and upper part of underground facilities are filled with non-contaminated fill material.

In this study, we analyzed site conceptual model of basement for overseas decommissioning NPPs. In general, underground structures and piping were considered establishing site conceptual model of basement. The results of this study are expected to be used as basic data for establishing site conceptual model for Kori unit 1 and Wolsong unit 1.

Keywords: Decommissioning, Site conceptual model, Basement, Buried piping, Embedded piping

Preliminary Analysis of the Characteristics of Waste Generation From Specific Decommissioning Activities

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As the number of nuclear power plants that have reached their design life increases, the demand for decommissioning nuclear power plants is increasing worldwide. In Korea, beginning with the decommissioning of Kori Unit 1 and Wolsong Unit 1, the demand for decommissioning is expected to increase. IAEA General Safety Requirement Part 6 states that a final decommissioning plan shall be prepared before the conduct of decommissioning actions and submitted to the regulatory body for approval. The final decommissioning plan should include the decommissioning strategy, the schedule, the waste management strategy, etc. In addition, the licensee shall demonstrate before their use that such methods are safe and can effectively achieve the desired results. Since Korea has not yet carried out the decommissioning of nuclear power plants, it is necessary to learn about the decommissioning experiences of other countries to prove the safety of the decommissioning method and establish strategies.

Electric Power Research Institute (EPRI) shares the summary of the decommissioning experience of nuclear power plants that have been decommissioned to provide lessons learned for future plants entering decommissioning. In addition, each nuclear power plant operator in the United States reports on the shipment of radioactive waste from nuclear power plants, liquid and gaseous effluents, and changes in the radioactive waste treatment system through the Annual Radioactive Effluent Release Report (ARERR). ARERR is recorded according to the format of Regulatory Guide 1.21, and solid waste shipment data are recorded by classifying them into 4 types. Since the change in radioactivity and volume for each type varies depending on the characteristics of the nuclear power plant and the decommissioning strategy, considerations for decommissioning work can be derived by analyzing the data. In addition, by using the data to derive the correlation between radionuclides, it can be used as an index to check whether decommissioning is well carried out when decommissioning nuclear power plants in the future.

This study analyzes the solid radioactive waste data of the U.S. Pressurized Water Reactor (PWR) nuclear power plant that has been decommissioned and analyzes information that can contribute to safe decommissioning by identifying the connection with decommissioning work. The results of this study are expected to help establish solid radioactive waste management strategies in decommissioning plans in countries with no decommissioning experience.

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Keywords: Decommissioning, Radioactive waste, Solid radioactive waste, Radionuclide analysis

A Review on the Management of Mixed Waste for Decommissioning of Nuclear Power Plants

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A large amount of radioactive waste is expected to be generated when nuclear power plants are decommissioned following the permanent shutdown of Kori Unit 1 (June 2017) and Wolsong Unit 1 (December 2019) in Korea. Currently, approval of the Final Decommissioning Plan (FDP) for Kori Unit 1 is in progress in accordance with relevant laws and regulations. In relation to the regulations, Section 9.5 (Mixed Waste Management) of the FDP Review Guidelines includes the types and characteristics of mixed waste expected to be generated during the decommissioning process. Therefore, a series of processes from generation to disposal are required.

Types of mixed waste that are expected to be generated include solid, liquid, and gas mixed waste, and most mixed waste is expected to be generated in solid form. Mixed waste is a waste that contains both radioactive and non-radioactive hazardous substances and contains explosive, flammable, corrosive, and epidemic substances, making it difficult to completely separate and dispose of them. When decommissioning a nuclear power plant, it is necessary to proactively determine whether recycling is possible and minimize the amount of mixed waste generated through non-hazardous waste management. Therefore, in this study, we reviewed mixed waste cases of nuclear power plants experienced abroad.

Mixed waste is expected to have an impact on the cost and schedule of nuclear power plant decommissioning in the future due to the absence of detailed procedures. When domestic conditions related to mixed waste disposal are established, we hope that this study will be helpful as a reference for management measures for mixed waste from domestic nuclear power plant decommissioning.

Keywords: Decommissioning, Radioactive waste, Mixed waste, Waste management

An Overview of Asbestos Abatement for Plant Decommissioning

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Pressures to retire and decommission the remaining coal fleet continue to mount as power generators worldwide transition to low-carbon or carbon-free energy sources. This study provides examples of relevant regulations specific to the United States, however, the planning and processes related to asbestos abatement as part of plant decommissioning and demolition discussed herein are applicable globally, and to nuclear power plants other than coal-fired power plants. The identification, abatement, handling, and disposal of asbestos can account for 10% or more of decommissioning costs, in addition to adding weeks or months to decommissioning schedules. Often, regulated materials such as asbestos are identified once decommissioning is underway and can lead to schedule delays and budget increases while the materials are removed. No facilities are exempt from asbestos regulations, regardless of the facility age or previous asbestos abatement activities performed. Locating, abating, handling, transporting, and disposing of asbestos within the regulatory guidelines is a critical task for any decommissioning project. This study will provide an overview of key considerations for planning and implementing asbestos abatement as part of plant decommissioning and demolition. Decommissioning projects benefit from developing and regularly updating a site-specific risk register to assist with identification and mitigation of risk. Each decommissioning team will need to assess the risk related to asbestos depending upon the level of confidence in the regulated materials inspection, the quantity of asbestos to be abated, the potential for impacts on schedule and budget, and other project-specific factors. The rules and regulations covering asbestos are varied and often requirements for activities such as decommissioning are contained within multiple areas. As discussed above, those working in countries other than the U.S. or on nuclear power plants facilities other than coal-fired power plants may have different or additional regulations related to asbestos identification, abatement, handling, storage, and disposal. Each decommissioning team will need to assess the risk related to asbestos depending upon the level of confidence in the regulated materials inspection, the quantity of asbestos to be abated, the potential for impacts on schedule and budget, and other project-specific factors. Abatement in some locations may require partial demolition to take place first to provide access, then the demolition is paused while abatement occurs. Some utilities prefer to utilize on-site landfills for asbestos disposal to decrease the total cost of abatement. Some sites already have permitted landfills that could submit a minor modification to allow disposal of asbestos, which is typically a much quicker process (months) than permitting and constructing a new landfill (years). If an on-site landfill were permitted and/or a new landfill permitted and constructed while the plant is operating, no delay in the decommissioning schedule would be incurred. In all fields, including asbestos abatement, leading practices for planning and execution are constantly changing as new methods that improve risk mitigation and work performance are developed. Some leading practices are required by regulations, such as having a certified asbestos supervisor on-site while abatement is performed, while others, like using viewing windows in containment areas so work is directly observable by the asbestos supervisor, are beneficial but optional. The laws and regulations governing asbestos identification and abatement vary in many countries. These activities can constitute a significant portion of the schedule and budget for plant decommissioning. Understanding the requirements, and how they align with larger project goals and objectives, is critical to project execution. Each company undergoing decommissioning will need to make many decisions based upon the considerations, while including the experience of internal and external decommissioning managers and other project personnel to define, plan, and implement the activities necessary to safely perform asbestos activities in compliance with applicable regulations.

Keywords: Decommissioning, NRCOC (non-radiological constituents of concern), Site assessment

Current Status of Detailed Design of Radioactive Waste Treatment Facilities in Preparation for Decommissioning of Nuclear Power Plants

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When decommissioning a nuclear power plant, unlike an operating nuclear power plant, a large amount of radioactive waste is generated. Decommissioning of nuclear power plants requires efficient and systematic treatment and management of radioactive waste generated in large quantities.

The purpose of constructing a decommissioned radioactive waste treatment facility is, first, to collect, classify, measure, decontaminate, and dispose of radioactive waste generated in large quantities during decommissioning; second, to secure the disposal period for radioactive waste through rapid radioactive waste treatment; and finally, minimizing the amount of radioactive waste generated and disposal costs.

Currently, Korea Hydro & Nuclear Power (KHNP) is pursuing detailed designs for related facilities to dispose of large amounts of radioactive waste generated during the decommissioning of Kori Unit 1. The treatment facility is scheduled to consist of four floors above ground and two floors underground.

For detailed design of the relevant treatment facility, a list of equipment to be built according to the amount of radioactive waste generated by level/type and detailed specifications for each equipment must be determined. To this end, through benchmarking and case studies of the design, operation, and licensing experiences of nuclear waste treatment facilities of domestic and foreign nuclear power-related organizations, we have derived a list of equipment scheduled to be built in the facility and are reviewing detailed specifications for each equipment. The list of equipment currently scheduled to be built at the facility will consist of cutting/measuring/decontamination/volume reduction equipment.

Once construction of the treatment facility is completed in the future, it is expected to play an important role in the treatment and management of radioactive waste generated in large quantities during the decommissioning of domestic commercial nuclear power plants.

Keywords: Decommissioning, Radioactive waste treatment facilities, Kori unit 1

Development of the Metal Incorporated Mesoporous Silica Adsorbents for Hydrogen Gas Separation

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Tritium (^3H) is radioactive isotope of hydrogen which emits beta ray with half-life of 12 years. Tritium can be undesirably occurred while operating heavy-water reactor, such as Wolsong Nuclear Power Plant (NPP). More specifically, the heavy water with deuterium (HDO or D_2O), used for the moderator, is converted as tritiated water (DTO or T_2O). Due to the its hazardous property as well as useful value, separation of tritium is necessary and currently, tritium removal facility (TRF) in Wolsong plays an important role. TRF separates the hydrogen isotope through catalytic exchange reaction and cryogenic distillation. Catalytic exchange reaction is basically converting tritiated water as gas phase, and further cryogenic distillation enriches the tritium via gas separation. Since the cryogenic distillation is extremely energy intensive, the alternate technique is required.

In this presentation, we introduce the development of the metal incorporated mesoporous silica adsorbents for hydrogen gas separation. To increase the adsorption capacity for hydrogen isotope, the metal incorporated mesoporous silica is newly synthesized via co-condensation method. Then, physicochemical and textural property is characterized by multiple techniques to elucidate materials. Then, non-radioactive gas separation is tested with single gas conditions such as H_2 and D_2 to estimate actual application.

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Keywords: Tritium, Gas separation, Tritiated water, Mesoporous silica, Adsorption

Analysis of Recycling Methods as a Solidification Agent for Radioactive Concrete Waste Generated During Nuclear Power Plant Decommissioning

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Korea's radioactive waste disposers and disposal sites are targeting 14,500 drums of decommissioning waste per nuclear power plant unit. However, according to previous studies, 80,000 drums of decommissioning waste is expected to be generated for the Kori unit 1 reactor, which is about to be decommissioned. Therefore, it is essential to reduce the amount of waste to improve the capacity of the disposal site and reduce the cost of decommissioning.

Since the amount of concrete waste generated during decommissioning is about 70% or more of the total waste volume, concrete waste disposal costs account for the largest proportion of decommissioning costs. In Korea, given the limitations of waste disposal volume and cost, it is essential to review technologies to reduce waste disposal costs by reducing waste volume and separating and recycling radioactive waste.

To this end, in Korea, a pilot scale equipment has been developed by a domestic research institute to recycle radioactive concrete waste into regenerated cement. In particular, the method of heating and grinding concrete waste at about 600°C, separating aggregate and fine powder, and then regenerating cement in the fine powder to manufacture the waste form is considered to be effective. In this case, the amount of fine powder in the aggregate is the most important, as it is difficult to secure the durability of the solidification agent as the amount of fine powder increases. In addition, the manufactured waste form must pass the standards (compressive strength ≥ 3.44 MPa, leachability index ≥ 6) for solidification requirements (compressive strength, thermal cycling, irradiation, immersion, and leaching tests).

Therefore, in this study, a large amount of concrete waste generated during the decommissioning of a nuclear power plant was targeted and reduced by heating grinding, and the fine powder was recycled as a solidification agent. If this technology is used in the future to confirm whether the domestic Waste Acceptance Criteria (WAC) is satisfied, it is thought to be a technology that can efficiently dispose of concrete waste generated during nuclear power plant decommissioning.

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Keywords: Decommissioning, Volume reduction, Concrete waste, Waste acceptance criteria

Characterization of the Mixed-Waste Generated for Decommissioning of Kori Unit 1

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The type of radioactive waste that may occur in the process of nuclear power plant dismantling can be classified into solid, liquid, gas, and mixed waste. The amount of these wastes must be defined in the Final Decommissioning Plan for approval of the licensing. Also, in the case of Mixed-waste, it is necessary to calculate the generation amount in order to store or disposal at a Radioactive Waste Storage Facility.

Based on the experience of the U.S (Connecticut Yankee) decommissioning nuclear power plant, the six types of mixed waste are expected to be generated during the decommissioning process of Kori Unit 1. The sources and causes of the mixed wastes are as follows :

- 1) Asbestos : insulation, ceiling materials, refractory materials, gaskets
- 2) Lead : shield (such as sheet type and brick type)
- 3) PCB (Poly Chlorinated Benzene) : epoxy paint, transformer, capacitor, cable insulation materials, fiberglass insulation materials
- 4) Mercury : thermostat
- 5) Oil : lubricating oil, cutting oil, hydraulic oil
- 6) Liquid scintillation : liquid scintillation counter

Mixed-waste does not generate a large amount compared to other decommissioned radioactive waste, but since disposal requirements or reduction measures have not been determined, it may need to be stored for a long period.

Therefore, it is necessary to accurately calculate the amount of mixed-waste generated and radioactivity from decommissioned nuclear power plants and develop reduction measures. In this study, we first predicted the mixed-waste generated from Kori Unit 1 and identified the cause of its generation.

Keywords: Decommissioning, Mixed-waste, Decontamination, Radioactive waste

Development of Radiation Shielding Equipment for Safe Handling of Large Concrete Drums Containing Radioactive Waste

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Prior to the decommissioning of permanent shut down nuclear power plants (Kori Unit 1), efforts to access the suitability of disposal of nuclear waste generated during operation and to develop technologies to meet the requirements.

For example, the development of technology to meet the requirements for accepting waste generated before a disposal site is built.

Considering the limited space and high stack ability of the waste storage warehouse of Kori Unit 1 and 2, it was judged that large concrete drums could not be operated and handled by crane-type handling equipment. Therefore, it needed to apply a forklift system considering the radiation characteristics and heavy handling.

In order to handle waste drums within the storage warehouse, which includes heavy concrete drums with radioactive waste, requires the acquisition of industrial and radiation safety technologies.

This study focuses on researching the development of shielding devices to minimize radiation exposure and industrial safety for operators within the radiation control area.

An appropriate shielding device was designed to minimize radiation exposure of workers and to enable work within the legal exposure limit by utilizing the radiation shielding ability of materials and the distance from radiation waste.

In order to determine the thickness of the shield materials, the surface dose rate of the large concrete drum at the time of creation was converted to the surface dose rate at the present time. A system that can be operated in a radiation environment was designed by integrating the calculated shielding materials with a forklift.

This research development is expected to contribute to minimizing operator radiation exposure and ensuring industrial safety within the workspace in future similar device development processes.

Keywords: Shielding materials, Large concrete drums, Handling of heavy materials, Industrial safety

A Preliminary Study on GUI Optimization for Decommissioning and Waste Management Software Based on 4th Industry Revolution

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Advances in computational techniques and equipment dramatically have led to improvements in existing technologies across engineering both national and global. Moreover, in the era of the fourth industry revolution (4IR), new technologies based on 4IR (i.e. big data, robotics, IoT, and artificial intelligence) that go beyond the limitations of conventional engineering technologies are spotlight due to their own advantages such as efficiency and economic feasibility, and so on.

Efforts which being led by governments and national institutes can be confirmed to apply 4IR technology in the field of nuclear power plant (NPP) decommissioning. It is of note that MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) workgroup have revised NUREG-1575 (MARSSIM) with the objective to include additional flexibility for improved and novel technologies and new survey methodologies to take technology development into account.

Furthermore, in December 2022, the Korean government (Ministry of Trade, Industry & Energy) officially announced that it would pursue the development and construction of 4IR technologies such as digital twin and robotics to strengthen the competitiveness of the NPP decommissioning industry in Korea. The Korean government proposed the goal of developing 4IR-based, ICT-utilized decommissioning technology, such as digital twin-based platform for decommissioning and in-situ robotics remote dismantling technology, which are one of the most widely developed technologies. Additionally, several foreign cases (e.g. FLYT from Norway, ACTS of U.S., etc.) reported decommissioning and radioactive waste management software development including waste tracking, which adopted the building information modeling (BIM) based on digital twin technology.

On the other hand, the most important development procedure of software is accurate and thorough analysis of requirements, especially in the decommissioning field with various stakeholders. In order to configure the main functions and ensure applicability of decommissioning software, it is essential to derive needs and considerations of stakeholders through in-depth and long-term requirements analysis. In particular, the development of GUI, which is prevalent in a wide array of software applications, is key to reflecting considerations derived through analysis of requirements.

Considering the 4IR and decommissioning policy and status of Korea, it can be expected that development of optimized digital twin-based software through requirements analysis will be necessary. According to governmental announcement in Korea and foreign cases of 4IR software development for decommissioning, this study aims to analyze the framework and basic requirements for developing the digital twin software for NPP decommissioning and suggest general functions with regard to optimization of GUI.

Acknowledgements

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Keywords: Decommissioning, Digital twin, Radioactive waste management, Forth industrial revolution

Radioactive Waste Management of Wolsong Unit 1 Reactor Decommissioning

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The radioactive structures removed during the dismantling process are packaged in a container and transported out of the building. Accordingly, the exposure effects that may occur in the process of managing radioactive waste including the process of cutting, packaging and transportation out of the radioactive structure, were evaluated.

During the operation, neutrons generated from the core are irradiated to the surrounding structures, and the radioactive level of each structure varies depending on the level, so they are packaged in different containers. It is assumed that radioactive waste generated during the dismantling process is packaged in a P3 container and an intermediate level radioactive waste in m Modular Shielded Container (MSC) container according to the radioactivity level of each structure. Through this, the number of containers required and the workers collective dose were evaluated.

Keywords: Dismantling, Wolsong unit 1, Container, P3, MSC

Characteristics of Liquid Radioactive Waste in the Case of Domestic Nuclear Facility Decommissioning

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The decommissioning of domestic nuclear facilities can be roughly divided into two main projects: the Seoul Research Reactor Units 1 and 2 decommissioning project, which commenced in 1997 and is currently ongoing at the Korea Atomic Energy Research Institute (KAERI), and the uranium conversion facility decommissioning project, which began in 2001 and concluded in 2012.

Liquid radioactive waste generated during the decommissioning of the research reactor can be broadly classified into two types. The first type comprises liquid radioactive waste that was generated during operation but remained even after the facility was permanently shut down. This waste mainly consisted of tank water and did not contain specific compounds, with radioactivity levels ranging from ND to $1.56\text{E}+05\text{ Bq/m}^3$ (gamma). The second type includes liquid radioactive waste generated during nuclear power plant decommissioning activities, such as: 1) Cooling water and tank water used during concrete cutting, which generally do not contain chemicals. 2) Decontamination waste liquids, including those from ultrasonic decontamination and steam spray decontamination, as well as mixed waste liquids of phosphoric acid and hydrochloric acid used in chemical decontamination, and laundry waste liquids (including surfactants).

Liquid radioactive waste generated during the decommissioning of the uranium conversion facility can also be broadly categorized into two types. The first type consists of liquid radioactive waste generated during operation but remaining even after the permanent shutdown of the facility. This includes waste generated during the operation of nuclear fuel manufacturing and conversion facilities, as well as uranium diluent (organic waste liquid containing T.B.P. and dodecane). The second type includes liquid radioactive waste generated during decommissioning activities, such as: 1) Decontamination waste liquids used in the facility dismantling and cutting process. 2) Steam spray decontamination and ultrasonic chemical decontamination waste liquids (using a 10wt% nitric acid solution). 3) Laundry waste liquids (which do not contain surfactants).

The characteristics of liquid radioactive waste generated during the decommissioning of the nuclear facility are based on the processing characteristics of liquid radioactive waste generated during the operation of the relevant decommissioning facility. Processes such as decontamination and dismantling must be selected with consideration of these characteristics.

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Keywords: Decommissioning, Liquid radioactive waste, Decontamination, Dismantling

Development of a High-Speed Radioactive Liquid Waste Treatment System Using Nanocatalysts

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In this study, we developed equipment to rapidly treat non-degradable radioactive liquid waste containing organic complexing agents generated from work such as replacement of old facilities and performance improvement during nuclear power plant operation and overhaul (O/H) period. The pilot-scale high-speed radioactive liquid waste treatment system using nanocatalysts largely consists of three stages of ‘nanocatalyst generation-precipitation-solid/liquid separation’.

The development equipment was of a mobile modular type and was minimized by placing each unit process in a 20ft container. This equipment conducting long-term experiments to derive optimal conditions in a variable external environment. 20 mM oxalic acid was used as a radioactive liquid waste simulation waste solution to evaluate the performance of the equipment and derive optimal treatment conditions. Copper (Cu) and nickel (Ni), which are nuclides mainly generated during decontamination, were selected as transition metal oxides for nanocatalyst creation and decomposition. In order to calculate the optimal conditions for oxalic acid decomposition by the selected nanocatalyst, the oxalic acid decomposition performance by reaction time was tested while changing the injection concentration of sodium persulfate and nanocatalyst metal oxide.

To date, it has been confirmed that approximately 95% of the treatment efficiency achieved through pilot equipment has been removed compared to before treatment, and experiments are continuing to achieve a treatment efficiency of over 98% by changing the condition variables to derive optimal conditions.

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Keywords: Radioactive liquid waste, Mobile and module type, Metal oxide, Nanocatalyst, High-speed treatment

Prospects of Temporary Storage and Final Disposal for Intermediate-Level Radioactive Waste From Decommissioned Nuclear Power Plants

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Kori unit 1 was permanently shutdown in June 2017 and is currently undergoing a licensing inquiry and response process to obtain approval for decommissioning. In the case of decommissioning a pressurized water reactor (PWR) such as Kori unit 1, the spent fuel (SF) classified at the potential high-level is generated, and in addition, the intermediate-level, the low-level, and the very low-level radioactive wastes are generated. Most of the reactor vessel internal (RVI) are the intermediate-level, and most of the overseas nuclear power plants that have been decommissioned or are in progress are considering disposal in deep geological repository along with the SF, i.e., the HLW. However, in the situation where the high-level disposal site is not in operation, RVI is managed by the independent spent fuel storage installation (ISFSI) along with the SF. In this paper, we will evaluate the components and radioactivity of RVI of decommissioned nuclear power plants around the world and discuss the current status of temporary storage and future final disposal plans.

Keywords: Intermediate-level, Reactor vessel internal, ISFSI, Final disposal

Treatment and Volume Reduction Plan for Kori Unit 1 Decommissioning Radioactive Waste By Using Induction Melter and Plasma Torch Melter

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The decommissioning of Korean NPP will be carried out starting with Kori unit 1, which was first built, and is expected to begin in earnest with the second permanent shutdown NPP of Wolsong unit 1. The Kori unit 1 is a Westinghouse PWR rated at a 587 MWe that began commercial operation on the Kori site in 1978 and was approved for permanent shutdown in June 2017. The progress of the Kori unit 1 decommissioning project consists of four phases. Currently, KHNP is conducting a licensing response with regulatory body for the final decommissioning plan (FDP), and the decommissioning will begin after obtaining approval from the Korean regulatory body. A radwaste treatment facility will be built and operated on the site to dispose of waste generated during the decommissioning. In case of waste generated from the decommissioning of the Kori unit 1, some parts of reactor vessel internal (RVI), along with SF, are classified as above the intermediate-level radioactive waste, and the remaining waste classifications are expected to be low-level radioactive waste (LLW), very low-level radioactive waste (VLLW), and clearance waste (CW). Most metals except for intermediate-level radioactive waste will be treated and reduced by Induction Melter (IM). In addition, organic waste and asbestos, which are difficult to dispose of, will be treated and reduced by Plasma Torch Melter (PTM) inside the radwaste treatment facility. Accordingly, the final amount of decommissioning radioactive waste is estimated to be less than 14,500 drums based on 200 liter drum.

Keywords: Kori unit 1, Decommissioning, Radwaste treatment facility, Induction melter, Plasma torch melter

Development of Remote Sampling Device for Radioactive Waste Filter of Nuclear Power Plant

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The radioactive waste filter sampling, which is currently being conducted in nuclear power plants, is carried out by an operator using tools such as a perforator to collect samples, making it difficult to collect samples due to the occurrence of a slip phenomenon during drilling at the representative sample point of the metal support plate. In order to solve this problem, a remote sample collection device that can remotely cut the metal surface of the circular metal cartridge waste filter into a circular shape and collect the glass fiber sample installed therein was developed in connection with the waste filter compression/packaging system. The waste filter remote sampling device consists of three main parts : 1) a filter fixing part that prevents the separation of the waste filter during sample collection, 2) a filter collection part for collecting the circular sample of the waste filter, and 3) a sample housing that collects and stores the collected sample. Through the development of such remote sampling devices, it is intended to reduce radiation exposure of workers and solve the problem of radioactive contamination diffusion due to the scattering of high-radioactive particles of filters.

Keywords: Spent filter, Sampling

Assessment of the NORM(TiO_2) Waste Disposal Option in Korea Industry

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Titanium dioxide pigments are widely used to manufacture paints and other coatings, plastics, and paper, and the related industries are continuously growing worldwide. In Korea, as the related industries are activated, a technical review is required to ensure the safety of the treatment and disposal of waste containing naturally occurring radioactive material (NORM). The risk of exposure NORM waste need to be identify and contril to protect worker and public.

This study presents and analysis of NORM (TiO_2) waste disposal option and risk assesement for used in general industry. In order to systematically evaluate the safety of the treatment and disposal of NORM waste generated at a titanium dioxide pigment manufacturer a NORM generating facilities in Korea. We identified the current status of the facility and analyzed the relevant standards and status of overseas facilities and field survey data. The safety assessment for treatment workers and the general public, including the status of generation, treatment, and disposal waste was evaluated using the RESRAD code. Sensitivity analysis were conducted on several input parameters related to the U and Th daughter radionuclides for on site resident and industrial worker scenarios. The scenario for the dose assesment was limited to landfill, and the dilution fraction for waste disposal, and the effects of changes in the thickness of the subsurface soil layer were analyzed, and a comparative assesment with overseas standards.

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Keywords: Safety analysis, Disspoal, Resident, Worker, TiO_2 , RESRAD

A Review on Recycling and Reuse of Materials From Decommissioning of Nuclear Facilities

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Throughout the world, significant volumes of scrap and waste materials consisting of mainly concrete and steel arise from decommissioning nuclear facilities. The inherent value of these materials and the need to reduce the volume of material destined for radioactive waste disposal facilities make their recovery through recycling and reuse not only prudent, but necessary. A large proportion of generated material during decommissioning would be only slightly contaminated and, if decontaminated and released, would have a high value on the scrap metal market. If release for recycling and reuse is not an available option, the material would have to be placed in low-level waste repositories for disposal. Therefore, a significant amount of material and waste generated from decommissioning should be considered for recycling and reuse. The aim of this study is to conduct an examination of the current state of recycling and reuse in the nuclear industry, identifying issues in recovering concrete, steel and other valuable materials comprising a large portion of the waste generated during the decommissioning of nuclear facilities. Also, this study is a review of the 20 years of decommissioning experience since the initial 1996 report on Recycling and Reuse of Metals (NEA, 1996) was prepared. The referenced report provides information and insights into the practicality and usefulness of release criteria from the perspective of organizations engaged in decommissioning activities, and compared these impacts with those associated with a proposed “tiered” regulatory regime that would allow large portions of these materials to be recycled and reused. The tiered system provides four basic options. First, Tier A: Material that is surface contaminated or only slightly activated metal would be decontaminated as possible and unconditionally released for reuse or melting. Second, Tier B: Material that is volume contaminated would be melted in a regulated environment to achieve decontamination, followed by metal recycle in commercial smelters or mills, and processing for use in consumer products (conditional clearance). Third, Tier C: Material containing short half-life products would be melted and fabricated in a controlled environment, and released for a specific initial industrial use (e.g. steel bridge). Only possible for contamination or activation by radionuclides with “short” half-lives (depending on the useful life of the product). Fourth, Tier D: Material that cannot be released from regulatory control will be recycled or reused in the nuclear industry (e.g. waste containers for final storage). Tier D does not involve release from regulatory control. The referenced report pointed out that the establishment of internationally agreed, unconditional release criteria is a critical step to filling the need for a consistent, internationally accepted standard. It was suggested that such criteria should be established in a manner that will encourage, rather than preclude, the future establishment of conditional release criteria. Consequently, shipment of material between countries, and even to other facilities within the same country, was found to be extremely difficult due to the absence of uniform criteria. Further complications were identified such as variations in the quality assurance requirements, sampling protocols, required instrumentation, and documenting practices. To this end, groups preparing to disband may refer to the “tiered” system of release criteria to facilitate discussion of appropriate conditional domestic release criteria.

Keywords: Decommissioning, Recycling, Reuse, Release criteria

A Review on Management of the U.S. Nuclear Plant Buried and Underground Piping

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As the number of years of operation of nuclear power plants, cases of damage due to aging of buried pipes are gradually increasing. This is an important issue facing nuclear power decommissioning. Deterioration of buried pipes causes contamination of groundwater and soil. However, in the case of buried piping, it is difficult to accurately find the leak and it is not easy to access, making it very difficult to manage. Currently, the management of buried piping in nuclear power plants is very limited, with leakage confirmed by monitoring the water level and pressure of the main system. In this paper, we analyze the United States' response and regulatory guidelines for buried piping and tanks of nuclear power plants and seek ways to apply them to domestic nuclear power plants.

Unlike above-ground pipes, buried pipes are simultaneously subject to corrosion from internal fluids, mechanical and chemical damage from external soil, and fatigue from heavy loads. When a leak occurs in a buried pipe, it is not only difficult to find the cause, but also difficult to access, making it very difficult to maintain it in a timely manner. Currently, buried piping health management involves identifying the location when a leak is confirmed and performing post-processing through temporary repairs or routine maintenance. Therefore, it is quite limited to identify signs of potential damage to buried pipes buried in soil or concrete and to perform soundness evaluation on them.

In the United States, at the request of regulatory agencies, the nuclear industry (Nuclear Energy Institute; NEI) uses the underground piping and tank management guidelines (NEI 09-14) to bury nuclear power plants and establishment of management plan for underground pipes and tanks and check implementation status.

In this study, Poor management of underground buried piping will eventually lead to contamination of underground water or soil, which will provide difficulties in decommissioning nuclear power plants in the future. As shown in the case of the United States, their management should be carried out systematically and systematically from the power plants in operation. The final dismantling of nuclear power will take place only when accurate and clear guidelines, transparent information disclosure, and regulations governing them match.

Keywords: Decommissioning nuclear, Buried pipe, Buried tank, Groundwater protection

Considerations for Root Causes and Contingency Management of Abnormal Situations During Decommissioning of Nuclear Facilities

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Abnormal events during decommissioning of nuclear facilities could not be removed, regardless of the efforts applied. This is especially the case in old facilities where documentation may have been lost and where modifications may have been carried out without updates to reports. As a result, when planning for decommissioning, it is prudent to assume that such problems will occur, and ensure that arrangements are in place to deal with them when they arise.

One of the most common root causes of unexpected events in decommissioning is the lack of detailed design information or missing records of modifications, maintenance issues and incidents during operation. Therefore, it is necessary that there is a need for a permanent contingency infrastructure to be available within a decommissioning organization. This must be the subject of a comprehensive contingency plan. This plan cannot be too detailed, but should include a contingency management policy which demonstrates that due attention is being paid to the likelihood of unexpected events and that these will be managed.

The contingency management of the unexpected events requires considerations of several factors such as organization, funding, planning, legislation and regulations, information, training, stakeholder involvement, and modifications to existing tasks. Preparation of organization management arrangements will be specific to the type of plant and the specific decommissioning activities. It may be necessary to release contingency funds from an identified contingency budget, depending on the seriousness of the event, to slow down or stop work on another project while funds are diverted to address the unexpected event. The planning arrangements in the contingency infrastructure are essentially developments of those arrangements that will already be in place for planning of decommissioning under normal circumstances. In the case of some unexpected events, particularly the more serious ones, it may be necessary to seek regulatory approval to carry out tasks that would be outside the normal scope of activities for a plant or the technical specifications in force. The information requirements for contingency planning are very important to prevent or mitigate unexpected occurrences. Obtaining this information will require the availability of access to search facilities that can contain drawings, reports and other technical information. Training of staff in all aspects of decommissioning and waste management should mean that even though an event is unexpected, the capability of the staff is able to deal with it. The adoption of an open and transparent communication policy requires advices from all stakeholders through regular meetings of the planned activities on the site. A comprehensive decommissioning plan will generally include a number of detailed project addressing specific aspects of decommissioning. Even though boundary conditions and other circumstances are not expected, it is good practice to include contingencies and to include new methods and techniques, procedures and provisions.

Keywords: Abnormal situation, Contingency, Decommissioning, Nuclear facility, Root cause

A Method for Calculating the Preliminary Removal Cost for the Decommissioning of Domestic Nuclear Power Plant

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In accordance with the completion of the design life of domestic nuclear power plants, preparations for nuclear power plant decommissioning are underway, and for the approval of decommissioning, a Final Decommissioning Plan (FDP) must be prepared in accordance with the relevant laws and regulations. According to the FDP's regulatory guidelines, in Chapter 2 of the decommissioning plan, project management, the cost required for decommissioning nuclear power facilities must be quantitatively evaluated in consideration of the facilities and site characteristics. The cost estimation methodology for domestic decommissioning nuclear power plants complies with the guidelines presented by AIF/NESP-036. In this guideline, the entire flow chart for cost estimate is presented, and the decommissioning cost category are classified into three categories: Activity-Dependent costs, Period-Dependent costs, and Collateral costs. In particular, since Activity-Dependent Costs are related to the amount of decommissioning waste, it takes considerable effort and time to collect the input data necessary for cost estimate from the initial data investigation. On the other hand, period-dependent and collateral costs such as project management and license, utility energy, taxes, tec., can be calculated relatively easily compared to activity-dependent costs. Therefore, in order to generate input data for cost estimate and to perform the evaluation properly, the calculation method for each cost item must be systematically established.

In this paper, we intend to establish and present a domestic Removal Cost estimate method using AIF/NESP-036 guidelines. In order to calculation Removal Cost, decommissioning work area are setting, decommissioning waste quantity evaluation, work difficulty factor evaluation, and unit cost factor should be established. Using this input data, we propose a Removal Cost estimate method, establish a domestic systematic Removal Cost estimate method, and build a foundation for applying the permanent shutdown and decommissioning nuclear power plants in the future.

Keywords: Decommissioning, Cost estimate, Activity-dependent costs, Removal cost

A Review of Influencing Factors for Selecting the Decommissioning Strategy

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The operation of Nuclear Power Plants (NPPs) will be extended or permanently shutdown when the scheduled design life expires, and the owner of the facility decides not to extend its operation, it will enter the stage of decommissioning. Usually, there is a certain period of transition after permanent shutdown, and preparations for decommissioning are made at this stage, including facility modifications for dismantling. However, activities to plan and prepare for decommissioning should be carried out by reflecting the basis of the decommissioning strategy. The strategies typically include immediate dismantling and deferred dismantling, and the overall plan and implementation of decommissioning project will be controlled by the selection of two strategies. Therefore, this study intends to review the influencing factors to be considered in selecting these decommissioning strategies.

The intention of the operator is crucial in selecting a decommissioning strategy, but there will be factors to consider in order to be implement the project according to the determined strategy. However, these influencing factors are not simple and must take into account in various aspects. As references to these, related matters can be found in GSR Part 6 Requirement 8 (Selecting a Decommissioning Strategy). IAEA NW-G-2.1 (Policies and Strategies for the Decommissioning of Nuclear and Radiological Facilities), IAEA SRS 50 (Decommissioning Strategies for Facilities Using Radioactive Material), and IAEA SSG-47 (Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities). In addition to referring to overseas guidelines, it is necessary to select a decommissioning strategy by reflecting domestic guidelines and circumstances.

In this study, we looked at the influencing factors suggested in the IAEA references. The influencing factors presented in references were listed, and although there are some factors that differ somewhat, it was confirmed that they had similar contexts overall. In addition, it showed the contents suggested by the guidelines of domestic regulatory authority and confirmed that they were consistent with the factors suggested by international standards. Although national policy and the will of the operator are important factors in each country, it will be possible only when the availability of waste management facilities, securing costs, securing dismantling technology, and other factors are considered in order to be viable in promoting the decommissioning project rather than considering only one fragmentary aspect.

Keywords: Decommissioning, Dismantling strategy, Immediate dismantling, Deferred dismantling

A Review of BIM Technology Application for Decommissioning NPP

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As cities have been formed for a long time, various solutions have been proposed in the construction sector to solve the problems caused by the aging of the city. Collaboration between participants is important to efficiently manage this project, such as large and complex structures and the management of vast amounts of information, and the collection, sharing, and management of information are important factors. In this way, BIM (Building Information Modeling) is being introduced as a way to efficiently and systematically manage various information and share information among participants.

BIM is a technology and process that integrates and manages information generated during the life cycle of a construction project from planning, design, construction, and maintenance stages based on a 3D model. It is a digital model that provides a reliable basis for decision-making during the facility life cycle and the work procedures for its creation.

In Korea, Wolsong unit 1 and Kori unit 1 are preparing for decommissioning. In general, a demolition process based on complete decommissioning is considered rather than a large-scale repair. Since the decommissioning of such old buildings is based on direct and physical damage through demolition equipment in the structures constituting the building, the frequency of collapse of the structure is higher than that of new construction, and when the structure collapses, large-scale collapse accidents and safety accidents can occur.

In this study, we analyzed BIM usability effective. This can minimize the interference between processes and errors schedules. And this decrease design changes and errors for workers and reduce the amount of waste generated economically. The results of this study are expected to be used as BIM for NPP decommissioning.

Keywords: Decommissioning nuclear, BIM (building information modeling), 3D modeling

Analysis of ALARA Evaluation Methodology for Overseas Decommissioning Nuclear Power Plant

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Kori unit 1 and Wolsong unit 1 permanent shutdown in 2017, 2019, respectively. In Korea, decommissioning of nuclear power plant (NPP) can be completed if the results of Final Status Survey (FSS) are less than Derived Concentration Guideline Level (DCGL). In terms of ALARA, if the need for remediation below DCGL is reasonable, the ALARA Action level can be set by analyzing the remediation cost and benefits of averted dose. Therefore, in this study, ALARA evaluation methodologies for overseas decommissioning NPP were analyzed. A number of NPPs have been completed decommissioning worldwide. We analyzed the cases of Yankee Rowe and Maine Yankee NPPs that have completed decommissioning in United States. The methodology of Appendix N of NUREG-1757 was used in Yankee Rowe NPP. If post-remediation survey results were lower than the screening level set by the NRC staff, it was considered to be optimized and no additional remediation was performed. The methodology of NUREG-1727 was used in Maine Yankee. Basement fill model (resident farmer scenario) was used to soil and standing building occupancy model (building occupancy scenario) was used to buildings. The remediation methods were selected soil excavation, scabbling, pressure water washing, wet and dry wiping, and grit blasting. Finally, the cost of applying each remediation method and the benefit of averted dose were compared. Both NPPs used the methodology presented in the NUREG. If the post-remediation survey results meet the screening level, Final Status Survey is performed, otherwise additional remediation or survey unit specific ALARA evaluation is performed. The results of this study will be used as basic data for establishing the ALARA action level before Final Status Survey.

Keywords: ALARA evaluation, ALARA action level, Remediation, Screening level, Final status survey

ALARA Strategy for Decommissioning Heavy Water Reactors

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ALARA plans for the entire process of preparation and dismantling should be established to ensure the radiation safety of workers when decommissioning calandria.

The ALARA goal is to minimize individual and group exposure doses by preventing internal and external exposure of workers by unplanned radiation. To this end, major items were derived and analyzed, including reduction/removal of radiation sources, prevention of the generation and diffusion of radioactive materials, work distance and radiation shielding, minimization of work time, utilization of remote automation tool, education and training, work plan establishment and safety evaluation, analysis of domestic and international experiences, radiation monitoring, and dose measurement.

Keywords: ALARA, Calandria, Exposure, Radiation safety

Analysis of Radiation Risk During PHWR Reactor Decommissioning

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The main radiation risk when calandria is dismantled is external exposure by gamma rays emitted from the radioactive structure being dismantled. And exposure by radioactive corrosion products deposited on the surface of the primary heat transfer system is also an important external exposure risk factor. It is mainly present in feeder pipes and steam generator tubes. Some corrosion products stick to the pressure tube and the inner surface of the end fitting, and the radioactivity of the corrosion products is estimated to be less than 0.1% of the total radioactivity when the reactor is dismantled.

In addition, there may be a risk of external exposure by tritium in the primary heat transport system and moderator system. In addition to tritium, C-14 is a potential internal exposure risk factor and should be mitigated by ventilation systems, HEPA filters, and wearing personal protection equipment to address radiation risks in the air.

Keywords: Radiation, Corrosion products, Primary heat transport system, Tritium, C-14

A Review on First Spent Fuel Disposal Facility: ONKALO Disposal Facility

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Recently, Finland announced that the world first spent nuclear fuel disposal facility will start operation in mid-2020's. The Onkalo spent nuclear fuel disposal facility is a deep geological disposal facility for the final disposal of spent nuclear fuel. It is close to Olkiluoto Nuclear Power Plant in Eurajoki, on the west coast of Finland. The disposal facility is about 250 km away from Helsinki. The construction and operation are responsible to Posiva, a company owned by two existing nuclear power operators (Fortum and TVO). About 100 disposal tunnels will be excavated in the facility. About 3,250 canisters will be disposed in the tunnel. Multi-barrier system is deployed for the safe disposal of the spent fuel. The spent fuel assembly will be placed in inner canister, composed of carbon steel. The inner canister is placed in the outer canister, composed of Cu. The outer canister will be placed in deposition tunnel. The deposition tunnel will be filled with buffer bentonite and backfill material.

In this study, systematic review on ONKALO disposal facility will be reported, including history of Finland waste management history, multi-barrier system and structure, encapsulation plant, etc.

Keywords: Spent fuel, Onkalo, Disposal, Posiva

A Study on the Building Demolition Plan and Guidelines

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Buildings that were built before are changing into new buildings at the end of their lives or with the development of the city. It reflects the trends of the times as time goes by, and due to safety problems of old buildings, new buildings are being built. Buildings that were built before are changed into new ones at the end of their lives or with the development of the city. According to this trend, the demand for decommissioning work has also increased rapidly, and all buildings are legislated to require a report of decommissioning in accordance with the Building Management Law enacted on May 1, 2020.

The laws and regulations for dismantling buildings are shown in Chapter 4 of the Building Management Law in the dismantling and destruction of buildings. The work of dismantling work should proceed in the following order: preliminary investigation, field investigation, selection of dismantling method, review of structural safety, preparation of dismantling plan, establishment of pollution prevention plan, designation of supervisor, performance of dismantling work, disposal of residues, and submission of completion report. do. The main tasks are as follows.

For Kori unit 1, the decommissioning plan is in the process of being reviewed for approval by regulatory agencies, and for Wolsong unit 1, the decommissioning plan is being publicly inspected by residents before submitting it to the regulatory agency. Even if the approval from these regulatory agencies is terminated, as seen in the above consideration, it appears that another approval for the dismantling of general buildings must be obtained. Since each licensing agency and the required documents are different. KHNP needs to process the necessary documents and permits from each agency.

Keywords: Building demolition, Statutes and regulations for demolition

A Study on the Safety Evaluation of Adjacent Units to be Considered in the Decommissioning of NPPs

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Two domestic reactors, Kori Unit 1 and Wolsong Unit 1, are preparing for decommission. The decommissioning plan for Gori unit 1 is under review for approval, and the decommissioning plan for Wolsong unit 1 is being publicly inspected. These NPP are adjacent to operational NPP that are currently generating power. And some facilities are shared with subsequent units. Accordingly, consider risk factors that find out how to prevent them.

The IAEA provides guidance for systematically assessing the safety of decommissioning operations and potential accident scenarios. IAEA Safety Guidance No. 77 is used to perform an assessment of occupational safety risks with key facility characteristics during operation and decommissioning. These characteristics highlight the safety implications of decommissioning.

To evaluate the risk of adjacent exhalation, five items were evaluated. This requires caution as it can have a fatal impact on adjacent units. These are the risk of fire, the risk of toxicity and hazardous substances, the importance of groundwater management, and the importance of vibration and structural safety management.

This assessment was conducted in accordance with the IAEA Safety Assessment for Decommissioning. The main purpose of the evaluation should be to ensure the safe operation of adjacent units by considering possible site conditions or changes. There may be risk factors in addition to the above, but the factors with the greatest risk are described. Apart from radiation safety, which is emphasized when decommissioning a nuclear power plant, the above matters must also be considered.

Keywords: Safety evaluation, Adjacent NPP risk, Decommission NPP

Review of the Media Considered at the Time of License Termination Phase of Overseas Decommissioning Nuclear Power Plants

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In the license termination stage of the decommissioning nuclear power plant, the Derived Concentration Guideline Level (DCGL) for the subject of deregulation (i.e., the media) must be satisfied below the allowable value to remove the regulation of the nuclear safety law. Each dismantled nuclear power plant may differ depending on the laws and regulations of each country for each media, and it is also complicated to set acceptance standards for them. The dose criteria for site release is simply expressed as 0.1 mSv/yr or 0.25 mSv/yr, but there are many factors to consider in deriving DCGL. Above all, it is necessary to evaluate the dose of the exposure from the residual radiation by considering the exposure pathways. Therefore, it will be necessary to review what contents and what considerations the previous overseas decommissioning nuclear power plants considered.

We selected Zion, Rancho Seco, Connecticut Yankee (CY), and Humboldt Bay as the target nuclear power plants to review the contents of overseas cases. In the case of soil, all four nuclear power plants were evaluated through the REARAD code in consideration of the exposure pathways through dose evaluation, and DCGL was derived as a result. In most cases of site reuse scenarios, the resident farmer scenario was applied, and in the case of Rancho Seco, the industrial worker scenario was applied. For buildings and structures, Rancho Seco, CY, and Humboldt Bay performed dose evaluation, and DCGL was derived respectively. Most of the applied scenario occupancy scenarios were considered, and Rancho Seco even considered the Renovation scenario. In the case of underground facilities, three nuclear power plants except Humboldt Bay performed dose evaluation, and buried pipe, embedded pipe, and concrete were evaluated. For groundwater, DCGL was evaluated through the RESRAD code for CY.

In this study, we investigated the contents of decommissioned nuclear power plants in the United States, which generally contain well-relevant data as an overseas cases. More specific studies will be conducted in the future based on this study on the media that can be considered in the license termination stage and the contents of dose evaluation for them. Therefore, the methodologies and contents carried out by overseas nuclear power plants for the deregulation work can be referred to as useful data for the decommissioning of domestic NPPs.

Keywords: Decommissioning, RESRAD, DCGL, Site reuse scenario

Current Status of License Response to the Final Decommissioning Plan for Kori Unit 1 Related to Management of Decommissioned Radioactive Waste

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The regulatory agency's license review of the final decommissioning plan, which is the licensing required for the decommissioning of Kori Unit 1, Korea's first commercial nuclear power plant, is in progress. The final decommissioning plan for Kori Unit 1, consisting of a total of 11 chapters, was prepared based on "Article 22 of the Enforcement Rules of the Nuclear Safety Act" and "Nuclear Safety and Security Commission Notice No. 2018-10" and submitted to the regulatory agency.

In particular, among the matters written in the final decommissioning plan for Kori Unit 1, inquiries from regulatory agency regarding "Chapter 9 Radioactive Waste Management" account for a large portion, and a response to the fourth inquiry is currently being written.

Key questions regarding Chapter 9 of the final decommissioning plan for Kori Unit 1 include the presentation of a comprehensive management plan for decommissioned radioactive waste, the presentation of basis for evaluation of the amount of solid radioactive waste generated and radioactivity, and matters related to the decommissioned radioactive waste treatment facility.

In order to prepare a response to this inquiry, we are currently operating a "Decommissioning License Response TF" and are pursuing "Comprehensive Design Services for the Decommissioning of Kori Unit 1". In addition, we are responding to inquiries by reflecting the needs of regulatory agency through continuous communication with regulatory agency. In the future, KHNP plans to actively pursue licensing response to achieve the goal of obtaining approval for the decommissioning of Kori Unit 1 by the end of 2024.

Keywords: Decommissioning, Radioactive waste management, License response, Kori unit 1

Development of Considerations for the Design of Wolsong Unit 1 Radioactive Waste Treatment Facility

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Wolsong unit 1, the first PHWR (Pressurized Heavy Water Reactor) in Korea, was permanent shut down in 2019. When dismantling a nuclear power plant, a large amount of radioactive waste is generated for a short period of time. In order for the decommissioning to proceed without delay, the treatment of radioactive waste must be properly carried out. Therefore, the detailed design is in progress to construct a radioactive waste treatment facility (RWTF) to treat the decommissioning waste of Wolsong unit 1. In this study, considerations for design of Wolsong unit 1 radioactive waste treatment facility were developed. It should be introduced optimal equipment that consider the available capacity for decommissioning radioactive waste disposal. The capacity of the low and intermediate-level (LILW) radioactive waste disposal facility is based on 14,500 drums per 1-unit decommissioning, so it should be satisfied. It should be possible to manage various types of radioactive waste. When decommissioning a nuclear power plant, various types of waste are generated, including structures, systems, and components, cables, and mixed waste. At the decommissioning site, dismantling is being carried out, most of the radioactive waste will be transported to a RWTF to treatment. It should be ensuring sufficient storage space including intermediate level waste storage in the RWTF. In the case of disposal facilities, not only decommissioning waste, but also radioactive waste from various facilities such as operating nuclear power plants, research facilities, and industrial companies must be received. Therefore, bottlenecks may occur in the process. In addition, in general, only low-level radioactive waste can be disposed of in a LILW disposal facility. Therefore, it is necessary to ensure sufficient space in the RWTF to temporarily store the waste. Liquid and gaseous radioactive waste must meet the effluent control limit. Liquid and gaseous radioactive wastes are generated in the process of radioactive waste treatment at the RWTF. In order to discharge them into the environment, they must meet the Nuclear Safety Act and environmental laws and regulations, the same as operating nuclear power plants. Detailed design is in progress to construct of RWTF. The considerations developed in this study will be utilized as the basis for the detailed design.

Keywords: Decommissioning, Design consideration, Radioactive waste treatment facility, Wolsong unit 1

6분과

방사선환경 및 안전 (ORAL)



Tritium Monitoring to Monitor Ocean Discharge of TEPCO Treated Water

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In March 2011, the accident at Tokyo Electric Power Company's (TEPCO's) Fukushima Daiichi Nuclear Power Plant (FDNPP) released many radioactive materials into the atmosphere and ocean. More than 10 years after the accident, various efforts are underway to decommission FDNPP. TEPCO uses an advanced liquid processing system (ALPS) to remove radioactivity, except tritium, from the cooling water used to stabilize the reactors and store it in tanks at the facility. Retreating the treated water accumulated on FDNPP site is necessary for decommissioning. The water that has been treated to remove the primary radioactive materials contains tritium, and the Japanese government has been discussing its treatment. The quantity of treated water stored at the facility has reached its capacity, and the Japanese government commenced discharging it into the sea in 2023. Under the Japanese domestic law, the limit for releasing tritiated water from nuclear power plants is 60,000 Bq L⁻¹; however, TEPCO's policy for actual releases is to dilute the water to 1,500 Bq L⁻¹ or less.

Seawater monitoring by several domestic agencies is ongoing to assess the impact of the discharge of treated water into the ocean, which began in August 2023, in the environment. Measurement of tritium concentration in seawater is being conducted by the Nuclear Regulation Authority, the Ministry of the Environment, Fukushima Prefecture, and TEPCO, with approximately 120 measurement points set up around a 100 km radius centered on the FDNPP. The frequency of monitoring is determined according to the distance of the treated water discharge works and ranges from once a day to once a quarter year.

JAEA is evaluating tritium data in these environments. Since a large amount of data needs to be processed quickly, a flowchart was created to enable classification of importance levels according to several predetermined criteria. The judgment criteria were 1) tritium concentration exceeds the range of fluctuation of past data, 2) tritium concentration is on an increasing trend, 3) tritium concentration exceeds the level of TEPCO's causal investigation, and 4) the five sampling data in the vicinity are elevated. In this presentation, the details of the environmental tritium monitoring and the experience of applying the data evaluation method will be presented.

Acknowledgements

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Keywords: Tritium, Treated water, Fukushima daiichi nuclear power plant accident, Seawater monitoring

A Statistical Approach to Determine Evaluation Criteria for Proficiency Testing in the Environmental Radioactivity Analysis

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The Korea Institute of Nuclear Safety (KINS) reinstated its proficiency testing (PT) program in 2014 to assess the proficiency of Nuclear Power Plant (NPP) operators in analyzing radionuclides in environmental samples. Performance statistics, including relative bias and p-values, are utilized to evaluate reported results, a method also employed in the International Atomic Energy Agency (IAEA) and KINS PT programs. Evaluation scores are determined by comparing performance statistics with predefined criteria, based on fitness for purpose or past PT rounds' expectations. This criterion, termed maximum acceptable relative bias (MARB) in IAEA PT, serves as a critical value in assessing the accuracy, precision, and trueness of reported results, with MARB values varying depending on the radioactivity concentration and analysis complexities for each radionuclide and PT program.

The MARB ranges for various radionuclides in spiked water samples, such as ^3H , ^{90}Sr , ^{137}Cs , ^{134}Cs and ^{241}Am in spiked water sample are 15~20%, 10~15%, 20%, 15~20%, 15~25%, 15~25% and 20~25%, respectively. Similarly, for soil samples, ^{238}U , ^{137}Cs and ^{40}K exhibit MARB ranges of 20~20%, 20~25%, and 20~25% based on past KINS PT evaluation data.

A statistical approach to determine MARB is proposed in ISO 13528 appendix E.8, setting MARB as three times the expected standard deviation derived from the regression equation for robust standard deviation based on the evaluation results from previous rounds of KINS PT. MARBs obtained from the ISO 13528 approach for the aforementioned radionuclides range from 7~18%, slightly lower than the current values due to participants' good performance. MARBs for other groups were also calculated using previous PT results from regional radioactivity monitoring stations and compared with those of NPP operators.

Keywords: Proficiency testing (PT), Maximum acceptable relative bias (MARB), ISO 13528, Robust standard deviation

Determination of Effective Source Geometry for *in Situ* Gamma-Ray Spectrometry at Underwater Sediment

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Significant amount of radionuclides were released after the accident at Fukushima Daiichi nuclear power plant accident. The radionuclides were deposited into ground surface and aquatic system, such as river, reservoir, and ocean. One of the most released nuclides, ^{137}Cs , was accumulated in the underwater sediment owing to its high affinity sorption in clay minerals in soil. The radioactivity of ^{137}Cs at the underwater sediment can be estimated by sampling soil and analysis in laboratory or conducting *in situ* measurement. We measured the radioactivity of ^{137}Cs up to the 30 cm depth at underwater sediment and measured gamma-ray spectrometry with NaI (TI) scintillator. And a Monte Carlo simulation was conducted to determine the effective source geometry of underwater sediment. We applied the measured activity to the simulation result to calculate expected count rates and it compared to the real data.

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Keywords: Underwater radiation measurement, In situ gamma-ray spectrometry, Radioactive cesium

Airborne Radiation Survey in Fukushima and Analysis Algorithms

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Airborne radiation monitoring (ARM) facilitates the swift assessment of extensively contaminated regions with minimal personnel, rendering it effective for urgent interventions. Dose distribution maps, created from aerial survey data, prove proficient in visualizing atmospheric dose rates and ground-level radioactive contamination. However, disparities in altitude can engender notable disparities between contamination information ascertained from airborne and ground-based sources. Therefore, implementing a calibration technique becomes imperative.

This study developed algorithms to predict ground contamination distribution and magnitudes incorporating airborne radiation monitoring data. One algorithm can be applied when contamination distribution changes are severe, and the other algorithm can be used in uniform topographical source.

Subsequent validation through the simulation data confirmed the effective prediction of the dose rates and distribution patterns. After the Fukushima nuclear power plant accident, KAERI (Korea Atomic Energy Research Institute) and JAEA (Japan Atomic Energy Agency) conducted several joint surveys of environmental radiation monitoring. Airborne survey and walk survey data near the FDNPP also used for validation. As a result, the predicted dose-rate distribution patterns and values accurately matched the real contamination.

Acknowledgements

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Keywords: Environmental radiation survey, Airborne radiation monitoring, Ground contamination, Analysis algorithm

Proposal for the Design Criteria of Decommissioning Radiation-Controlled Area to Optimize Occupational Exposure in Decommissioning Nuclear Power Plants

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As of 2024, Korea is operating a total of 25 nuclear power plants (NPP), including 22 pressurized light reactors (PWRs) and 3 pressurized heavy water reactors (PHWRs). Particularly, Kori Unit 1 was permanently shut down in 2017 and is scheduled to be decommissioned for the first time in Korea. Preparations are underway through various methods to decommissioning Kori Unit 1, which includes radiological characterization, the most important factor.

In a decommissioning NPP, various radiological hazards generated during the operation exist in each area. The operating NPP operates radiation-controlled areas (RCAs) to protect workers from these radiological hazards and to provide appropriate protective measures. The RCA design criteria presented in the current summary of the final decommissioning plan (FDP) for Kori Unit 1 are the same as the final safety analysis report (FSAR) for Kori Unit 1. However, unlike operating NPPs, decommissioning NPPs are actually dismantled. Therefore, RCAs should be set in consideration of the radiological environment of each zone, and the exposure dose of workers should be minimized.

This study analyzed the RCAs in the FSAR of the NPPs, including both operating and permanently shut-down NPPs, to propose design criteria for the RCA in decommissioning NPPs. The RCA of PWRs was divided into 4, 5, 6, and 8 areas, and design dose rates were given for each area. The RCA of PHWRs was divided into 6 areas, and the design dose rates were presented along with detailed explanations for each area. In addition, the RCA of PHWRs includes the derived air concentration (DAC) considering internal exposure by tritium. The RCAs of each NPP were identified, and then the design criteria for the decommissioning radiation-controlled areas (DRCAs) were proposed, such as external radiation exposure area, air contamination area, and surface contamination area.

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Keywords: ALARA, Decommissioning, Radiation controlled area, Radiation worker, Final safety analysis report, Nuclear power plant

Assessing Non-Radiological Risks of Lead in Gyeongju Radioactive Waste Disposal Facility

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In nuclear facilities, mixed wastes, which is a radioactive waste including hazardous material, are continually generated for various purpose such as radiation shielding and radioisotopes analysis. In accordance with the waste acceptance criteria for Gyeongju's low to intermediate level radioactive waste disposal facilities, the removal of chemical hazards within mixed waste is necessary for acceptance. However, considering that radioactive waste disposal facilities have a higher level of safety in design and management criteria compared to hazardous waste disposal facilities, the disposal of mixed waste, which presents relatively low non-radiological risks, through the radioactive waste disposal facility can reduce the time and cost burdens associated with decontamination processes.

In this study, a simplified safety assessment designed for the silo-type disposal facilities for soil was conducted for the lead, which is commonly used for the shielding purpose, with the Contaminated Land Exposure Assessment (CLEA) model, material transport modeling program GoldSim, and Ecolego. With GoldSim and Ecolego, site-specific data concerning engineering barriers and natural barriers at the Gyeongju disposal facility was utilized as an input, thus an understanding of Pb mobility was obtained. CLEA model was used to define the permissible limits within radioactive waste inventory, ensuring not to exceed generic assessment criteria of soil.

This research demonstrates the feasibility of assessing the non-radiological risks of mixed waste, considering radioactive waste disposal site characteristics, by utilizing verified safety assessment methods for radioactive/non-radioactive wastes.

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Keywords: Mixed waste, Disposal facility, Safety assessment, Lead, CLEA

Development of Sampling Device to Evaluate Radiation Deterioration Characteristic of NPP Bioshield Concrete and Environmental Analysis of Sampling Operations

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NPP bioshield concrete ensures safety of people and structures by isolating radioactive materials from the outside, and by protecting the internal structure from the external environment. The exterior of NPP bioshield concrete deteriorates non-radiologically due to environmental factors, and the interior deteriorates radiologically due to neutrons and gamma rays. In particular, during the radiation deterioration process, neutrons can react with the crystal atoms of the concrete structure and cause the crystalline substance to become non-crystallized, resulting in serious deterioration of the concrete. To prove structural safety, a concrete core sampling device is needed to evaluate the radiation deterioration characteristics of bioshield concrete for long-term NPP.

When coring radioactively contaminated bioshield, concrete dust is inevitably generated. Since dust causes the spread of radioactive contamination, a wet coring method is necessary to prevent dust scattering. The water used in the wet coring process can also spread contamination and should be recovered along with the dust. To address this issue, we developed a concrete core sampling device that can recover dust along with water during the core sampling process, in order to safely evaluate the radiological deterioration characteristics of bioshield concrete.

Through a literature search, we determined the concrete mixing ratio likely to have been used at the Kori NPP unit 1. We produced concrete test specimens based on this ratio in order to evaluate the performance of the concrete core sampling device that we developed. For the test specimen, we performed tests on device installation time, coring time, and degree of water leakage to evaluate the device's performance. During the performance verification process of the device, we confirmed that no dust was generated, and there was no water leakage from the surface of the coring area.

In order to evaluate the radiation deterioration characteristics of nuclear power plant concrete, even among nuclear power plants that have been in operation for a long time, the nuclear power plant to be decommissioned must be evaluated with the nuclear reactor core removed. Since this is not currently feasible, we analyzed the operating conditions and the surrounding geometry of the Kori Unit 1 reactor geometrically. And the MCNP code was used to simulate the bioshield's radioactive contamination distribution sample collection environment.

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Keywords: Bioshield, Deterioration, Deterioration of concrete, Core sampling device

Machine Learning Based Pre-Feasibility Test for Beta Nuclides Identification

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The pre-feasibility test for beta nuclides identification was performed by using machine learning method. Due to the risk of internal exposure by beta radiation, the systems for detecting and monitoring it are of great necessity for the safety of workers of nuclear facilities, making the continuous real-time monitoring of beta radiation important. This study evaluated the identification probability of beta nuclide performing machine learning based on the detection probability of each nuclide by energy using spectrum data.

The pre-feasibility test was conducted divided into simulation using Monte Carlo N Particle 6 (MCNP6) code and experiment, where the detector of CAM 450 model with a good resolution among the passively-implanted planar silicon (PIPS) detectors was employed for the experiment. For machine learning, random forest and gradient boosting models were used and accuracy for identification was evaluated. ^{60}Co and ^{204}Tl sources were used in MCNP simulation. The machine learning confirmed the identifiability of simulation data, which showed an identification probability of 100%, followed by machine learning implement using the experimental data. $^{90}\text{Sr}/^{90}\text{Y}$ and ^{137}Cs sources were added and used to check whether more nuclides could be identified. Both machine learning models were found to have 100% identification probability for experimental data, too.

The present study representing 100% nuclide identification probability for a single nuclide environment showed it could provide the basis for the application to in-situ real-time monitoring and rapid tracking of nuclide sources in the multi-nuclides environment, indicating it would greatly contribute to the enhancement of radiation environment safety.

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Keywords: Machine learning, Random forest, Gradient boosting, Beta nuclides, Identification

On-Site Demonstration of a Fast Compton/Coded-Aperture Hybrid Gamma Imaging System for Detection and Localization of Radioactive Materials

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Gamma imaging devices capable of detection and localization of radioactive materials can be effectively used in various places including radwaste management facilities, nuclear facilities, nuclear medicine facilities, and airports and harbors. Although several hand-held devices have been developed, their application is seriously limited by low imaging sensitivity due to small sensor size. To address this limitation, a very fast Compton/code-aperture hybrid gamma imaging system, called Fast Gamma Imager (FGI), has been developed, boasting imaging efficiency at least dozens of times greater than that of conventional handheld devices. In addition, the FGI shows a good imaging resolution ($< 10^\circ$ angular resolution) over a broad energy range (59.6 keV – few MeV). The present study evaluated the FGI's performance and usability through on-site demonstrations. The first demonstration involved locating misplaced radiopharmaceuticals or radioactive waste at Hanyang University Hospital. The FGI accurately localized a heavily shielded ^{131}I (364 keV) radiopharmaceutical (896 μCi) in less than 1 seconds at 1.5 m distance, and a syringe containing 3 μCi of ^{131}I in just 9 seconds at 1 m distance. Furthermore, a radioactive waste container contaminated with the isotope $^{99\text{m}}\text{Tc}$ (140 keV), presumed to be in tens of microcuries range, was instantly localized from 2 m away. The second demonstration was localization of a radioactive substance hidden in a cargo at Incheon Airport Customs. The FGI successfully identified a cargo containing the industrial source ^{137}Cs , which contributed an additional dose rate of 0.033 $\mu\text{Sv/h}$ above the background dose rate, among plenty of cargos on the inspection table in just 1.6 seconds. Even in the conditions where the source contributed lower dose rate of 0.001 $\mu\text{Sv/h}$ above the background dose rate, location of the source was identified in 100 seconds, demonstrating FGI's high sensitivity. In a further test at the Customs, the FGI successfully localized a cargo containing a 240 μCi ^{137}Cs source moving along a conveyer belt with 0.15 m/sec speed, showcasing the FGI's exceptional imaging speed and real-time monitoring capability. Throughout these on-site demonstrations, the FGI demonstrated its versatility in various operational settings, offering new perspectives on utilizing radiation imaging for enhanced radiation safety.

Acknowledgements

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Keywords: Fast gamma imager, High sensitivity radiation detector, Hybrid imaging, Radiation safety

6분과

방사선환경 및 안전 (POSTER)



Analysis of the Criteria for Distinguishing Between Internal Exposure and External Contamination When Using Standing Type Whole Body Counter

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Internal exposure occurs when radioactive material enters the body. If there is a concern that internal exposure may occur, worker protection must be taken for radiation protection. Internal exposure monitoring is performed in three steps: (1) measuring the radioactivity of residual/external excretions in the body, (2) calculating the intake using the radioactivity measurements and the metabolic model provided by the ICRP, and (3) performing a dose assessment based on the intake of radioactive material. The first of these three steps, the measurement of radioactivity, is categorized into direct measurement (in-vivo radio-bioassay), which measures radioactivity in organs and tissues of the body, and indirect measurement (in-vitro radio-bioassay), which measures radioactivity in excreta.

In general, direct measurements are preferred due to the advantages of rapid measurement and high accuracy. However, it has the disadvantage of not being able to distinguish whether the measured radioactivity value is due to internal deposition or external contamination, so a criterion that can clearly distinguish between the two (internal and external contamination) is required for accurate dose assessment.

In this study, we examined the criteria for determining whether the measurement is due to internal exposure or external contamination by using the ratio of the upper and lower detector coefficients of the whole body counter when performing body radiation measurements with a standing whole body counter. In the case of internal exposures, the ratio of the upper and lower detector coefficients was within about a factor of two to three because the major organs are located in the upper body. In the case of external contamination, certain detectors had unusually high counts. Because the face, hands, and feet were selected as the main areas of contamination, and these locations are skewed toward one location.

The results of this study can be utilized in internal exposure inspection and quality assurance procedures for internal exposure radiation dose, and can be applied in cases where it is necessary to quickly distinguish between internal exposure and external contaminants (such as emergency radiation disaster situations).

Keywords: In vivo radio-bioassay, Internal exposure, Whole body counter (WBC)

A Method for Re-Evaluating the Release of Radioactive Materials Using Measured Radiation Doses After a Nuclear Power Plant Accident

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The licensee of Nuclear Power Plants in Korea, operate S-REDAP (Smart-Radiological Emergency Dose Assessment Program) for prompt emergency response to nuclear accidents. S-REDAP predicts the release amount of radioactive materials based on the circumstances and type of accidents and evaluates the dispersion of radioactive materials using meteorological forecast data. It also evaluates the radiation dose due to dispersed radioactive materials and ultimately recommends appropriate protective actions for public. Generally, accurately predicting the amount of radioactive materials released depending on the circumstances of a nuclear accident is almost impossible, so it is evaluated considering conservative situations. Consequently, the contaminated area by radioactive materials and the expected radiation dose are likely to be conservatively assessed and leading to unnecessary protective actions for residents in some cases. To minimize these situations, S-REDAP also has a function to re-evaluate the initially predicted release of radioactive materials using the measured radiation doses after an accident. For re-evaluating the release of radioactive materials, information on the doses based on the initial prediction from S-REDAP and the measured radiation doses should be available. In addition, the predicted doses and the measured doses should correspond to the same locations. It is expected that environmental radiation monitoring systems near the nuclear power plant site will be utilized for re-evaluating the release of materials shortly after the accident. When predicted and measured doses for multiple locations are collected, the correction factor that minimizes the difference between two kind doses for each location can be calculated using the least squares method. Finally, the correction factor is multiplied to release amount of radioactive materials and dose re-evaluation and modified resident protection actions are recommended using the corrected release amount.

Keywords: Nuclear power plant accident, S-REDAP, Emergency response, Radionuclide release

Feasibility Assessment of TLD Dose Algorithm Based on Artificial Neural Networks

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Spent nuclear fuels stored in pool are expected to be saturated in 2032 based on the Kori NPP in South Korea. As a result, Korea Hydro & Nuclear Power (KHNP) aims to operate a dry cask storage facility in 2030. When transporting spent fuel for operation of a dry cask storage facility, workers may be exposed to radiation. Accurate dose assessment of personal exposure in such situations is an important factor of radiation protection for safety of radiation workers and compliance with established dose limits.

In this study, we propose a more precise and simplified approach to constructing dose assessment algorithms by utilizing the computational power of artificial neural networks (ANNs) to automate and improve the algorithm generation process of TLDs among personal dosimeters. The developed algorithm is adept at identifying the mixtures of different radiation fields and calculating the mixing ratio. We also show a new method for building a dataset for training and testing the artificial intelligence models that comprise the algorithm. To evaluate the developed algorithm, tests were performed on its ability to discriminate radiation fields and calculate mixing ratios based on the performance test categories presented in “ANSI N.13 11-1993”.

Therefore, we aim to apply artificial intelligence to the dose assessment algorithm, reducing the resources required for algorithm design and improving the accuracy of readouts. It is expected to help develop the field of radiation safety management.

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Keywords: Artificial neural network, Dose assessment algorithm, Thermal luminescence dosimeter, Radiation protection, Dry cask transportation

Review on the Method for Reduction of Radiation Dose by Radioactive Carbon in the Gaseous Effluent of NPPs

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The number of nuclear power plants (NPPs) has been increasing in Korea. The operation period of the plants is also to be extended with continuous operation policy to meet the electric energy demand in the future. The radiation dose of the residents around the NPPs needs to be minimized through more thorough management of gaseous effluent to secure public acceptance on the continuous operation by minimizing the environmental impact of NPPs although it is assessed to remain within legal standards.

The nuclides with a high contribution to radiation dose by gaseous effluent of NPPs are radioactive carbon (C-14) that is mainly created with the activation of O-17 and N-14 by neutrons in the nuclear reactor core and eliminated to a large extent by the activated carbon of air cleaning unit (ACU) in the plant ventilation system and by the ion exchange resin in the reactor coolant purification system.

The radioactive carbon (C-14) in gaseous effluent is mostly released in ¹⁴CO₂ form through the ventilation system of spent fuel building and spent resin storage building although the emission is below the plant design value.

We reviewed on the methods that can control thoroughly the creation of C-14 and capture more effectively the radioactive carbon created or desorbed in unexpected situation during plant operation.

Keywords: Gaseous effluent, Radioactive carbon, Environmental impact, Reduction of radiation dose

A Study on Public Dose Assessment About Recycling of Radioactive Steel Waste Through Regulatory Clearance

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The operation and decommissioning of nuclear facilities can generate large amounts of contaminated or cleared radioactive steel waste. In particular, the cleared steel, which is disposed of outside the scope of radiation safety regulations, is commonly recycled, affecting the public. In general, workers who directly handle contaminated or cleared scraps during the recycling process are expected to have a higher dose to the public because they spend relatively less time exposed to radionuclides. Therefore, scenarios with potential exposure to the general public should be discussed and evaluated in detail. However, the development of RESRAD-RECYCLE, which program is commonly used for the evaluation of domestic for the regulatory clearance (self-disposal) processes, has been discontinued since 2005. Accordingly, it is limited in adding scenarios suitable for domestic use or improving existing scenarios. In order to solve the problem, it is necessary to develop a dose assessment program of steel waste with self-disposal through recycling.

In this study, assessments of 11 different scenarios for public were performed. The dose calculation to the general public from the self-disposal of steel waste were evaluated by assuming the recycling of 100 tons of steel waste. The steel wastes are assumed with a concentration of 1 Bq/g homogeneously contaminated with ^{60}Co , ^{137}Cs , and ^{238}U , respectively. The assessment results were compared with RESRAD-RECYCLE and with IAEA Safety Series 111-P-1.1, which is used to derive international standards for clearance levels. In comparison of results, the number of overlapping scenarios with this study was 10 in RESRAD-RECYCLE and 6 in the IAEA document. There was also one scenario in which RESRAD-RECYCLE and the IAEA document did not overlap each other. In several scenarios, the dose results for each nuclide were similar, while in other scenarios there were significant differences. These results are likely due to differences in detailed assumptions and input parameters.

The dose results for each nuclide still differ from the results of the other assessment methods such as the IAEA document and RESRAD-RECYCLE. Further research on scenario configuration and input parameters are needed for future studies. Exposure doses should also be considered for recycling process workers. In particular, worker scenarios have the potential for breathing and intake, so it is necessary to update dose conversion factors (DCFs). It is expected that these additional studies will lead to a replacement of RESRAD-RECYCLE in the future.

Keywords: Clearance, Recycling, Steel waste, Dose assessment

Performance Test Results for Whole Body Counters in Nuclear Power Plants

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The internal exposure due to the radionuclide intake is likely to be occurred in Nuclear Power Plants (NPPs). To monitor the internal exposure for workers in NPPs, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) operates Whole Body Counters (WBCs) according to plant procedures. The WBC measured radionuclide which emits high-penetration radiation, such as photon, retained within the worker's body. The internal exposure dose can be evaluated using the concentration of radionuclide remaining in the body measured using WBC. Therefore, the performance of WBC measurements significantly influences the reliability of internal exposure dose assessments. The KHNP conduct performance test of WBCs in all NPPs for every year to confirm the performance of the equipment. The performance test is conducted using unknown sources and the participants of the comparison test submit the radionuclide and activity of the unknown sources measured by WBC as a result. The performance indicator and criteria for WBC recommended in the American National Standards Institute (ANSI) N13.30 report published in 2011 are applied. The performance indicator is Root Mean Squared Error (RMSE) and criteria is 0.25 or less. The results of performance test performed in 2023 for all WBC is meet the ANSI N13.30 criteria. And the RMSE values are confirmed from 0.03 to 0.17.

Keywords: Internal exposure, Whole body counter, Performance test, ANSI N13.30

Comparison Test of Reading Capability for Personal Dosimeters in Nuclear Power Plants

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In nuclear power plants (NPPs), individual dosimeters are used to manage radiation exposure from external sources for workers. In domestic NPPs, Thermo-luminescent dosimeters (TLDs) are utilized as personal dosimeters, and dosimeter reader facilities are registered and operated on a plant. According to Article 115, Paragraph 1 of the Nuclear Safety Act Enforcement Decree, dosimeter reader facilities must undergo performance tests conducted by regulatory agencies. Currently, before the regulatory agency's performance test, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) conduct annual comparison test for TLD reader capability between NPPs. According to the Nuclear Safety Act of Korea, there are eight categories for the performance criteria of TLD reader capability. The performance indicators and criteria for each category (bias (≤ 0.35), standard deviation (≤ 0.35), or acceptable limit (≤ 0.50)) reflect the American National Standards Institute (ANSI) N13.11 (1983), adopted in the ordinance. The assessment results of all participating dosimeter reader facilities in 2023 were found to meet all categories compared to performance criteria. The domestic NPP TLD reader capability comparison test described in this paper is conducted as part of worker dose assessment quality management and is expected to contribute to improving the reliability of evaluation results.

Keywords: External exposure, Thermo-luminescent dosimeters, Performance test, ANSI N13.11

Changes According to the Application of the Cover Layer in Dose Evaluation for Site Release of Nuclear Power Plant

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The ultimate purpose of decommissioning of nuclear power plants is to deregulate the site and remaining buildings after the end of decommissioning. At this time, it is necessary to prove that the concentration of radioactivity remaining for the contaminated medium is below the criteria. Usually, the Derived Concentration Guideline Level (DCGL) is derived by the licensees, and the site characteristic parameters is reflected in the regulatory criteria presented as doses (0.1 mSv/yr). In the case of the site, the exposure pathways from contamination to the exposed person are considered and the computer program RESRAD code is used. In the application of the site characteristic parameter, the cover layer on the top of the contaminated layer can be conservatively evaluated as zero. However, in reality, an cover layer of soil or concrete material can be applied. Therefore, this study aims to evaluate and analyze the effects of cover layers and radionuclides.

There are several factors that need to be determined in order to use RESRAD code. It will be the determination of the input value reflecting the site reuse scenario, the selection of deterministic/probabilistic evaluation methods, the selection of radionuclide, and the site characteristics evaluation. We applied the input parameter values according to the deterministic method in previous studies and the resident farmer scenario was considered. In the case of radionuclides, representative Co-60, Sr-90, Ni-63, and Cs-137 were selected. For the cover layer material, two general soil and concrete were selected, and the density of the material was 1.5 g/cm^3 and 2.4 g/cm^3 , respectively. It was evaluated by changing the thickness of the cover layer, and the applied values were evaluated by changing 0, 1, 3, and 5 cm.

In this study, the dose was evaluated for each cover layer thickness for soil and concrete for the four radionuclides, and a comparative graph was presented. Only applying the cover layer itself showed a significant dose reduction effect, and it was found that there was a significant effect as the thickness increased. In the case of concrete material, as the density increases, the attenuation effect of the dose increases, and in the case of Co-60, which has a high contribution to the external exposure dose, the attenuation effect was large. The results showed that the effect of this cover layer is significant when the site is reused in actual decommissioning stage.

Keywords: Decommissioning, RESRAD, DCGL, Site reuse

Estimation and Comparison of Worker Dose According to Decommissioning Schedule

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It is expected that the effect on the dose from major tasks with high radioactive contamination will be significant in predicting the dose of workers in dismantling work. These tasks must be described in the Final Decommissioning Plan (FDP), which is a licensing document, before the dismantling can be carried out, and their validity must be ensured. In Korea, Kori Unit 1 has been permanently shut down, and a FDP has been submitted to the regulatory body and is undergoing licensing review. In order to show that the dose prediction evaluation of dismantling workers is valid, it will be necessary to compare the overall dose value with that of overseas cases. Therefore, this study intends to compare the worker dose values estimated by overseas nuclear power plants by applying a proposal for the schedule of major tasks.

In the previous study, we initially set the performance of the high radiation dose work to 2025 in the case of Kori-1, but the possibility of delay until 2032 was left open due to changes in the actual plan. Therefore, if the predicted dose values are considered to have been evaluated as values generated by sequential delays from 2025 to 2032, the decay factor was applied for comparison with overseas cases. The decay correction factor was based on the assumption that major work takes place seven years after the permanent shut down of overseas Zorita nuclear power plants, where dose evaluation can be relatively standard. If Kori Unit 1's major dismantling works can be performed in 2025, it will be carried out eight years after permanent shut down, so the decay correction of about one year should be corrected for comparison with Zorita. Through this process, normalized doses were derived accordingly if the main work schedule changed from 2025 to 2032. In addition, the decay correction factor of Co-60 was considered as a representative nuclide, but the dose was also derived when the major nuclides had different corrections for Sr-90, Cs-237 and Eu-152. In addition, based on the decay correction factor of Co-60, the generalized dose values in 2025 were compared with overseas cases (CY, Zorita, MY, Rancho, SONGS 1, Zion 1, and Zion 2).

In this study, we derived a generalized value considering the decay correction factor for comparing the results predicted by the worker dose for dismantling with overseas cases. Comparing with normalized values with overseas cases is expected to be used as useful data in the future. However, it is important to understand that the main radiation work is the result for the target year that performs the high radiation work and the difference in year from the criteria year for overseas cases is compared.

Keywords: Decommissioning, Dismantling, Dose estimation, Decay correction factor

Neural Network-based Radiation Dose Calculator Program for Inhomogeneous Exposure Situations

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When working in a radiation-controlled area, safety controls must be in place to monitor exposed doses to protect workers from harm from radiation. In general, the dose to the worker is assessed by reading the personal dosimeter and multiplying it by a dose conversion coefficient, but in situations where the radiation field is inhomogeneous, this may under- or overestimate the exposure dose. The dose conversion coefficient can be calculated by Monte Carlo computer simulation for specific source distribution situation, but depending on the geometry, the calculation can be time consuming and difficult for non-experts to perform. In this study, a deep neural network model was applied to calculate dose values similar to Monte Carlo simulations. First, we developed a model that can predict the whole-body dose of a human phantom according to the position of the source. A space up to 1 m from the center of the phantom in both the + and - directions of the x, y, and z axes was set as the space of interest, and about 55,000 location samples were selected at 5 cm or 10 cm intervals in the space. For each location, the whole-body dose to the human phantom for a 1 MeV photon emitted in the isotropic direction was calculated as nps 10^{7-8} . From the output file, the source location information and whole body dose information were extracted to create a dataset, and trained the DNN model with 90% of the data, excluding 10% of the test data. The trained DNN model produces results similar to the Monte Carlo simulation with an error within about 3%. Using a whole-body dose prediction model for point sources, a program was developed to calculate whole-body doses when the source type is a straight line, square, or cube. After entering the type and size of the source, and entering the number of samples, a set number of point locations are sampled to be evenly distributed over the source area or space. By averaging the dose values for each location, a dose similar to the Monte Carlo simulation results can be calculated. The relative position of the sources to the human phantom can also be visualized using the 3D viewer. This program is expected to allow users who do not have the simulation code or basic knowledge of simulation to quickly and easily calculate human dose in external exposure scenarios. If a dosimeter dose prediction model is developed and added to the program, it will also contribute to the rapid calculation of the worker's dose from dosimeter readings.

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Keywords: Radiation dose assessment, External exposure, Monte carlo simulation, Deep neural network

Derivation of Time Response Curve of Radiation Survey Meter Using Cs-137 Standard Radiation Field

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Organizations that use radiation generating device must perform safety management in accordance with Nuclear Safety Act and measure the radiation dose rate to confirm suitability of radiation shielding. These organizations and Korea Institute of Nuclear Safety use commercially available radiation survey meters for the radiation safety management and regulatory inspection. Generally, the time response of these survey meter is about several seconds. Therefore, if the radiation emission time of the radiation generating device is short (less than a few seconds), measured dose rates may be underestimated. Accordingly, there is a need to analyze limitation for time-response of survey meters, and prepare method to measure appropriately considering characteristics of survey meters and radiation field of users. In this study, we derived time response curve of commercial radiation survey meters, using Cs-137 standard gamma radiation field. We conducted survey for status of radiation survey meters used in Korea, through business agents for radiation safety management. And we selected a total of 13 survey meters as the target in this study, considering survey meters most commonly used, manufactured domestically, and used by KINS. We measured time response of rise and fall for radiation dose for each survey meters, using Cs-137 radiation field of the Korean Association for the Radiation Application. The time response were measured under 3 dose rate: 1.21, 12.1, and 363 $\mu\text{Sv/h}$, and were repeated 5 times under same conditions. Overall, the time for complete rise and fall for each survey meters was mostly between 20~30 seconds at dose rate of 1.21 $\mu\text{Sv/h}$, and between 5~10 seconds at 12.1 $\mu\text{Sv/h}$, and between 3~6 seconds at 363 $\mu\text{Sv/h}$. The indication values of some survey meters were reset to background level at certain point during fall, or indication values changed significantly. Therefore, organizations using radiation generating device must select appropriate survey meters that can increase reliability of measurement, considering emission time and dose rate level of radiation generating device. Or, survey meters need to be exposed to the radiation, during sufficient reaction time. The result of this study can be used as fundamental results for effective regulatory management of facilities using radiation generating device.

Keywords: Radiation survey meter, Radiation generating device, Radiation safety, Time response, Radiation field

Safety Monitoring and Strengthening Measures for Building Decommissioning Work

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Despite the establishment of a dismantling construction system, the collapse of a hotel in Nakwon-dong, Seoul (January 2017), the collapse of a hotel in Jamwon-dong (July 2019), and the apartment collapse in Gwangju (June 2021) continue to occur. As a result of safety inspections at demolition construction sites across the country, a total of 153 violations were detected at 73 sites out of 210 sites. This is a matter of poor implementation of the system, such as poor preparation of the decommissioning plan, construction that is different from the decommissioning plan, and neglect of the duties of the decommissioning supervisor. This requires measures to prevent recurrence in order to strengthen safety management. Nuclear power plants preparing for decommissioning must also consider these problems and prepare safety management measures.

Educational improvement measures are required. Education to prevent safety accidents is the most basic measure. Prevent safety accidents by discovering potential risks and developing emergency response skills. In addition, the qualification system for decommissioning work must be improved to train companies that professionally perform decommissioning work.

In this study, we looked at the current status of human and material damage occurring during the actual dismantling of buildings, and looked into improvement measures in three aspects to prevent safety accidents during decommissioning construction. Many technical and institutional improvements have been made, and there are more guidelines to be followed. We must be aware of this and manage the decommissioning of complex and complex buildings, such as the decommissioning of nuclear power plants, safely and systematically.

Keywords: Building decommissioning, Safety monitor

International Studies on Extracting Dissolved Resources From Seawater

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Seawater contains energy resources that can be used for over 60,000 years, but efficient extraction technologies are currently lacking. In this study, we reviewed overseas research cases regarding methods for extracting seawater dissolved resources.

1. Radiation-Induced Graft Polymerization (RIGP) Method

When applying the radiation-induced grafting method, the Degree of Graft (DOG) varies depending on radiation factors (dose, dose rate) and grafting reaction conditions (e.g., atmosphere during RIGP process, residence time, temperature). Additionally, DOG is influenced by the ratio of supplied monomers and copolymers, and monomers such as acrylic acid (*AA*), methacrylic acid (*MA*), and glycidyl methacrylate (*GMA*) are evaluated to affect adsorption performance. Materials such as fibers exposed to radiation can be variedly used to achieve higher DOG, and there is a research case where ultra-high molecular weight polyethylene (UHMWPE) was used as a support and *UHMWPE-g-(PAO-co-PAA)* was synthesized by grafting *AN* and *AA*. Such adsorbents, using *GMA* and *MA* as grafting monomers on UHMWPE fibers exposed to radiation, achieved higher DOG.

2. Atom Transfer Radical Polymerization (ATRP) Method

ATRP method using catalysts is a controlled radical polymerization (CRP) technique, and ATRP allows control of monomers compared to free radical generation in RIGP, especially known to be suitable for controlled polymerization of *AN*. Saito et al. developed an adsorbent *PE-g-PVBC-g-(PAO-Co-PAA)-b-PAA* in the ATRP system by imparting hydrophilicity with *PtBA* through extension of graft polymer chains. Brown developed adsorbents using various shapes of chlorinated polyethylene (*CPE*) and polypropylene (*CPP*) fibers through the ATRP method, confirming higher adsorption performance by polymerizing *AN* and *t-BA*.

3. MOF, COF and PAF

Wang et al. developed adsorbents using UiO-66 metal-organic frameworks (MOF) for extracting seawater dissolved resources. Compared to MOFs, covalent-organic frameworks (COFs) with robust structures due to covalent bonding are advantageous for accommodating ligands for seawater dissolved resource adsorption. Additionally, porous aromatic frameworks (PAFs) have high versatility due to flexible molecular design, programmable pore structures, and high surface areas.

In addition to the methods described above, there are also electrochemical methods and photocatalytic methods for extracting dissolved resources from seawater. However, these methods have the drawback of being limited in usage. Through overseas research experiences, it is believed that more efficient development of adsorbents for extracting dissolved resources from seawater can be achieved.

Keywords: Extract, Seawater, Resource, RIGP, ATRP, Adsorbent

Industrial Safety Management Cases for Decommissioning of Nuclear Facilities

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Industrial safety system of a nuclear facility encompasses a multifaceted approach that begins with a rigorous initial review process. This process is fundamentally anchored in strict compliance with the extant laws and regulations governing nuclear safety and environmental protection.

Through this essential process, organizations must set industrial safety objectives that are aligned with the unique characteristics and changes of the work environment. It is then crucial to carefully develop and implement detailed and strategic plans and processes aimed at achieving sustainable safety improvements, ensuring a safer workplace for all employees.

During the decommissioning of nuclear facilities, it's critical to empower managers with the authority to take appropriate actions in case of unexpected situations. To achieve this, a comprehensive response plan is required, which includes organization, funding, planning, legislation and regulation, information, training, stakeholder involvement, and modifications to existing programs. These measures need to be prepared in advance to effectively respond to the complexities and unforeseen circumstances of the decommissioning work.

In this research, we have conducted an in-depth examination of the industrial safety guidelines applicable to nuclear facilities and analyzed real-world response cases to unforeseen events encountered during the decommissioning of nuclear reactors. The ultimate goal of this study is to leverage these insights to significantly enhance safety management practices in future nuclear decommissioning projects, thereby making a substantial contribution to improving the overall safety of these complex and critical processes.

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Keywords: Nuclear facility, Industrial safety, Decommissioning, Safety management

Analysis of Industrial Safety Accident Occurrences in Decommissioning Sites of Nuclear Facility

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Decommissioning of a nuclear power plant requires a variety of equipment and a significant amount of labor over a long period and involves both radiological and non-radiological hazardous tasks simultaneously.

Protecting workers from the numerous risks associated with the decommissioning process is important. These include not only the direct radiological hazards of handling and dismantling nuclear materials and infrastructure but also a range of non-radiological risks from general industrial hazards associated with heavy machinery to exposure to chemical substances and physical injuries. To effectively address these challenges, it is essential to implement comprehensive risk mitigation measures that encompass both radiological and non-radiological aspects.

Furthermore, new safety regulations are required to address the physical environmental changes that occur during the nuclear decommissioning process, which are not covered by existing nuclear safety regulations. To achieve this, a comprehensive investigation of accident cases that have occurred in nuclear facilities both domestically and internationally is necessary. Based on an analysis of accident cases, this study aims to identify potential gaps between existing safety practices and regulations and the required safety management measures during decommissioning, with the goal of proposing methods to enhance workers' safety.

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Keywords: Decommissioning, Industrial safety, Accident, Hazard

Introduction of Environmental Radiation Investigation Method Around Nuclear Power Plant in Dismantled Nuclear Power Plant

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Kori Unit 1 and Wolsong Unit 1 were permanently suspended in 2017 and 2019, respectively. It was decided that the relevant nuclear power plant would not be reused but be demolished. Various large structures in nuclear facilities will be removed in stages, considering safety. The dismantling of nuclear power plants is not the reverse order of construction. The process of dismantling is a battle against radiation safety, which has high uncertainty, unlike construction that goes according to a consistent process. In addition to safe dismantling, it is also important to consider because of the dismantling of nuclear power plants. It is a radiation effect on the surrounding environment.

Most of the residents were worried about the adverse effects of radiation that residents would suffer from the dismantling and ordered measures. Therefore, scientific evaluation and methodology for environmental radiation are needed. The purpose of investigating environmental radiation around nuclear power plants is to ensure the health and safety of residents and to prevent radioactive contamination of the environment by confirming that the radiation dose received by residents around the power plant due to dismantling of nuclear power plants is sufficiently low within the annual dose limit.

This study examines the legal content, scope of preparation, and technical matters of environmental radiation evaluation at home and abroad, and suggests what to prepare for the upcoming domestic dismantling nuclear power plants by considering the monitored nuclear species considered by overseas nuclear power plants.

Keywords: Environmental radiation, Nuclear dismantling, Kori unit 1, Wolsong unit 1

Conceptual Designs of the Automated Preconcentration and Purification System for Determination of Radionuclides in Seawater

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Radionuclides of interest ($^{134/135/137}\text{Cs}$, $^{89/90}\text{Sr}$ and $^{238/239/240}\text{Pu}$) in seawater, released from the normal operation or the accidental operation of nuclear facilities, were regularly monitored for environmental assessment. The isotopic ratios within and between the radionuclides of interest, potentially useful for long-term tracking radionuclides transport in seawater, can be used as a tool of understanding how radionuclides are transported from different nuclear production source terms and distributed in the ocean.

Radionuclides of interest in seawater were determined using radioanalytical methods, including liquid scintillation counting (or gas-flow proportional counting) for $^{89/90}\text{Sr}$, alpha-ray spectrometry (or mass spectrometry) for $^{238/239/240}\text{Pu}$ and gamma-ray spectrometry (or ultra-high sensitive mass spectrometry) for $^{134/135/137}\text{Cs}$.

Because of the extremely low concentration of the radionuclides in seawater, preconcentration and purification of the radionuclides are essential from the seawater matrix prior to instrumental measurements. Several techniques such as cation/anion exchange chromatography, extraction chromatography and coprecipitation have been used to isolate the radionuclides from large volume seawater samples.

In this study, the conceptual designs of an automated preconcentration and purification system were described for determination of radionuclides at trace levels found in seawater discharged from the various nuclear facilities.

Acknowledgements

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Keywords: Conceptual design, Automated preconcentration and purification, Radionuclides, Seawater

Emitters' Contributions to Background Concentrations of Radioxenon in the Northern Hemisphere

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There are many radioxenon emitters around the world from which routinely released radioxenon spread into the atmosphere under their normal operation conditions. Almost 200 of Nuclear Power Plants (NPPs), few tens of Nuclear Research Reactors (NRRs), and more than 10 of Medical Isotope Production Facilities (MIPFs) are operating nowadays and most are in the Northern Hemisphere. Recently the routine emission amount of radioxenon from each of worldwide nuclear facilities (NFs) were reported in detail for the year 2014.

Usually, the measurements at the International Monitoring System (IMS) stations operated by the Comprehensive nuclear Test-Ban-Treaty Organization (CTBTO) are continuously affected by radioxenon background. We carried out atmospheric dispersion simulations on routinely released radioxenon in the Northern Hemisphere up to 6 months (June to November) for the year 2014 using the Lagrangian Atmospheric Dose Assessment System (LADAS) with Unified Model Global Data Assimilation and Prediction System (UM-GDAPS) data produced by the Korea Meteorological Administration (KMA) including all of known emission sources which were operated in 2014.

Comparison of our simulation results with the measured data at four IMS stations, such as CAX17 (St. John's, Canada), DEX33 (Freiburg, Germany), SEX63 (Stockholm, Sweden), USX75 (Charlottesville, USA), most frequently affected by industrial radioxenon, showed excellent agreement, then we can analyze and reveal individual emitters' contribution to radioxenon background at each IMS station.

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Keywords: Radioxenon, Background concentration, Routine emission, Atmospheric transport and dispersion model, LADAS, UM-GDAPS

Reflections on Radiation Protection Cooperation in Northeast Asia Based on the Analysis of the Cases of the Convention on Environmental Impact Assessment in a Transboundary Context Related to Nuclear Activities

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Environmental protection regarding the use of nuclear energy cannot be limited to the impacts of radiation alone; the impacts and damages to the real world environment, including humans, are complex and often difficult to distinguish strictly between radiological and non-radiological impacts. We are also faced with the practical necessity of considering the impacts of nuclear activities not only within their territory under their jurisdiction, but also to a certain extent transboundary context. One of the positive initiatives is the Convention on Environmental Impact Assessment in a Transboundary Context (Espoo Convention), which was adopted in 1991 at the initiative of the United Nations Economic Commission for Europe (UNCEC).

This study analyzed the main contents and procedures of the Espoo Convention and the application of its implementation to nuclear activities. Also, the representative cases in European countries involving nuclear activities considered by the Espoo Implementation Committee were analyzed. The results highlight important environmental impact assessment issues that are currently being addressed internationally in relation to nuclear activities. It also found that by implementing the Espoo Convention, States Parties are paying for cooperation on radiation protection on a voluntary and proactive basis. In addition, they are strengthening radiation protection in the long term, and building confidence both domestically and internationally.

The findings of this study may have implications for the adoption of proactive radiation risks prevention measures in the form of environmental impact assessment agreements in recent situations in Northeast Asia, where China is constructing or already operating dozens of reactors along the coastline of the sea shared with South Korea, and for the development of international cooperation for radiological protection.

Keywords: Environmental impact assessment, Espoo convention, Transboundary, Radiation protection

Development of Automatic Nuclide Analysis Capability for Airborne Radiation Monitoring

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In the event of a radioactive release from a nuclear power plant, a prompt emergency response requires to analyze the behavior and estimate the emission of radioactive materials dispersed in the atmosphere. Since these radioactive materials spread under the influence of wind, real-time measurement and analysis are essential. If it is possible to measure radiation doses and analyze nuclide information in real-time, it can greatly assist in early response and the establishment of thorough disaster prevention.

Currently, airborne radiation measurement using unmanned aerial vehicles (UAVs) faces difficulties for real-time dose measurement and nuclide identification due to various issues such as payload limitations of onboard instrumentation, communication data loss, and power shortages. Furthermore, existing software was only applicable to single-channel detectors and could not utilize multi-channel applications. Additionally, after verifying the energy spectrum, users are disadvantaged by having to manually identify and analyze the energy peaks of radioactive nuclides of interest.

To improve these, this study developed software that can control multi-channel applications and automatically identify the energy peaks of radioactive nuclides of interest. The developed software applies an algorithm that automatically performs spectrum smoothing, peak identification, peak centroid determination, FWHM determination, and ROI setting from the measured energy spectrum, thereby allowing for automatic identification of energy peaks for radioactive nuclides of interest. Furthermore, the inherent Nuclide Library in the software can be utilized to estimate radioactive nuclides. By utilizing this function, users' analysis efficiency can be increased and it is expected that radiation doses and nuclide information of airborne radiation can be measured and analyzed in real-time.

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Keywords: Energy spectrum, Spectroscopy, Multi-channel radiation detector, Airborne radiation, Automatic peak analysis

Storm Surge Simulation Near Nuclear Power Plants Using ADCIRC and FVCOM Models

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Recent research has shown that the intensity of typhoons is increasing owing to global climate change, leading to a higher frequency of typhoons passing through the southern coast of Korea. This may increase the vulnerability of coastal areas to storm surges. Nuclear power plants are evaluated for disaster preparedness based on the most severe typhoons recorded. The Korea Hydrographic and Oceanographic Agency utilizes frequency central pressure derived from historical typhoons and artificially generated typhoon movement paths for evaluation. Coastal hazard maps were developed using ADCIRC, a 2-dimensional depth-integrated model. However, to accurately simulate the rapid changes in storm surges caused by severe typhoons, implementing a 3-dimensional model is necessary. Moreover, the absence of tidal gauge stations near nuclear power plants necessitates precise simulation of storm surge heights.

This study employed the ADCIRC and FVCOM models to analyze the reproducibility of storm surge heights based on different meteorological field types. This study highlights the characteristics of storm surge heights on the southwestern coast of Korea. Since 2002, the Joint Typhoon Warning Center has provided data on the radius, maximum wind speed, and central pressure at 34, 50, and 64 knots. The Japan Meteorological Agency offers a simulated JMA-MSM meteorological field covering the Yellow Sea, East Sea, and coast of Japan. An equivalent numerical simulation system based on the grid system outlined in a previous study was applied to assess the reproducibility of the 2-dimensional ADCIRC model and 3-dimensional FVCOM model using these meteorological data. Forty-one typhoons that impacted Korea between 2002 and 2021 were identified and compared with the observed storm surge heights at a tidal gauge station on the southwestern coast of Korea. The Root Mean Square Error (RMSE) is calculated to evaluate the accuracy of the simulation results. Consequently, the accuracy of the ADCIRC and FVCOM models improved to an RMSE of 0.16 and 0.12 m, respectively. Furthermore, the temporal pattern of storm surge height changes demonstrated a credible level of agreement when the FVCOM model with the JMA-MSM meteorological field was applied. Based on these findings, this study aimed to describe the characteristics of storm surge heights on the southwest coast and the maximum surge heights near nuclear power plants.

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Keywords: ADCIRC, FVCOM, Storm surge, Meteorological field, Nuclear power plant

Prediction of Typhoon Path Occurring in the Northwest Pacific Using Artificial Intelligence

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In a previous research, a stochastic synthetic typhoon was generated using a tropical cyclone risk model. However, a problem arose where typhoons undergoing landfall were incorrectly dissipated owing to the attenuation parameters. To address this issue, a probability density function was calculated based on historical typhoons to determine the movement path of typhoons that made landfall along the southeastern coast of China, allowing them to decay or resurface according to the calculated probability. Additionally, appropriate attenuation parameters were applied, improving the representation of synthetic typhoons passing through the southwestern coast of Korea compared with the observed values. Most synthetic typhoons landing on Korea's southern coast and Japan's eastern coast tend to dissipate rapidly. Applying suitable central pressure attenuation parameters to this model is challenging because of the limited number of historical typhoons that have made landfall, which makes analysis and application difficult. To enhance this model, further research utilizing artificial intelligence to predict typhoon movement paths is necessary.

This study utilized the best-track typhoon movement paths provided by the Joint Typhoon Warning Center from 1951-2022 as training data for artificial intelligence. Of the 1,881 typhoons, 1,721 were used for training, excluding those with movement paths shorter than 24 h. Eighty-five percent of these typhoons served as the training dataset, and the remaining were used to compare the forecasts with historical typhoons. The long short-term memory technique was employed using the Python library TensorFlow, with the typhoon's movement speed, direction, latitude, and longitude updated every six hours as input data. The model features two hidden layers and uses the tanh activation function. The learned model's Root Mean Square Error was 0.132° , and the predicted movement paths of the typhoons qualitatively matched the best-track data well. This study provides a foundation for generating extreme typhoons that could impact nuclear power plants and analyzing and evaluating the risks associated with storm surges and wave overtopping. Future research will use artificial intelligence to predict the maximum wind speed radius of typhoons and the radius of each wind speed in each quadrant.

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Keywords: Typhoon path, Best track, Artificial intelligence, Nuclear power plant

Wave Overtopping Simulation in Busan Marine City Using Smoothed Particle Hydrodynamics Model

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Flooding damage occurred in Busan's Marine City, located on the southern coast, owing to strong winds and wave overtopping. This incident resulted from Typhoons Maemi (TY0314), Chaba (TY1618), and Hinnamnor (TY2211) passing through the Korea Strait or making landfall on the southern coast, causing wave overtopping on several occasions. In a previous study aimed at predicting such typhoons, inundation from wave overtopping was forecasted by dynamically integrating the EurOtop module with the ADCIRC+SWAN model. However, a limitation is associated with the size of the finite element discretized grid applied; the total overtopping volume is small, leading to a determination of the dry state when the water level is lower than the $H_{\min}$ parameter, according to the wet-dry processing technique. Moreover, the overtopping volume is susceptible to the γ_f reduction coefficient in the EurOtop empirical equation, indicating the necessity for an alternative solution. In addition, situations where seawater overtopping occurs, as in nuclear power plants, may lead to accidents. Thus, although this may result in reduced speed, an additional method is required to analyze the behavior of seawater particles closely.

In previous research, several quantitative verifications have been conducted by applying the dual-SPH model through mathematical model experiments in a wave tank. This study aimed to reproduce the actual overtopping wave flooding that occurred in Busan's Marine City using the SPH method. The SPH model area was set to include the Marine City's land and bathymetry constructed in previous research. The boundary conditions of the SPH model were based on the significant wave height, period, and wave direction, which varied with time and were numerically simulated during Typhoon Hinnamnor using the ADCIRC+SWAN model. The SPH model was numerically simulated for 10 min at the peak significant wave height. The wave generation equation proposed by Madsen was used for a piston paddle. The initial distance between particles, ΔP , was set at 0.6 m, and the total number of fluid particles was approximately 15.66 million. Parallel computation was conducted using 16,384 CUDA cores of the NVIDIA GeForce TRX 4090 graphics card and required approximately 2 h. The numerically simulated results were qualitatively compared with inundation depths presented in previous studies, and the initial findings were considered satisfactory.

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Keywords: SPH, Wave overtopping, Marine city, Parallel computation, CUDA core

Development of Domestic Accident Scenarios for Maritime Transportation of Spent Nuclear Fuel

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In the case of transporting spent nuclear fuel, overland transport route presents various factors that contribute to a relative increase in radiological risks. Moreover, considering the difficulty in mass transportation and concerns regarding public exposure and security, maritime transport routes are more likely to be preferred. To evaluate risks for the maritime transportation, it is necessary to build accident scenarios covering all accidents that may occur along the transport routes. This study has developed the domestic accident scenarios specific to maritime transport of spent nuclear fuel for radiological risk assessment.

This study conducted an analysis of previous research conducted both domestically and internationally to evaluate the risks associated with the transport by maritime, and investigates the feasibility of incorporating domestic statistical data into the accident scenarios provided in these studies. Domestic maritime accident statistics from both the Korea Maritime Transportation Safety Authority and the Korea Maritime Safety Tribunal were analyzed and utilized for this purpose. As a result, the scenarios specific to domestic maritime accidents were developed through the analysis of judgements issued by the Korea Maritime Safety Tribunal. The applicability of radiological risk assessment methodology in conjunction with the 'GIS risk analysis' provided by the Korea Maritime Transportation Safety Authority was also examined.

The results of this study will be used as crucial data for regulatory verification research on spent nuclear fuel transportation risk assessment. This can also be utilized as fundamental data for the development of relevant technical standards and regulatory guides.

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Keywords: Spent nuclear fuel, Maritime transport, Accident scenarios, Risk assessment

Construction of Carboxylic Acid Functionalized Covalent Organic Frameworks for Uranium Removal

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Covalent organic frameworks (COFs) represent a novel class of crystalline porous materials comprised of constituent organic building units. Due to their high surface area, abundant coordination sites, and chemical and thermal stability, COFs have gained widespread attention in the field of water pollution remediation. In this study, carboxylic acid-functionalized COFs were successfully synthesized using a three-component one-pot reaction and subsequently employed for the removal of uranium under simulated solar light conditions. Comprehensive characterization of these COFs was carried out using techniques such as X-ray diffractometry, X-ray photoelectron spectroscopy, and Fourier-transform infrared spectrophotometry. Batch experiments were conducted to investigate the adsorption-photoreduction process of uranium by COFs. The results from physicochemical characterizations and removal experiments indicated that the removal efficiency was significantly enhanced by light exposure and was contingent upon the functional groups present on the COF surface. The mechanisms of uranium removal by COFs were discussed through experiments conducted under various conditions, including pHs, chemical scavengers, and additional characterizations of COFs before and after the reaction. This study will provide new ideas for uranium removal, as well as serve as a reference for the construction of COF structures with adjustable and efficient photocatalytic performance.

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Keywords: Covalent organic framework, Adsorption, Photocatalytic reduction, Uranium

Criticality Safety Evaluation of UO_2 Power Storage for LEU+ Preparedness

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As part of the project to expand the processing facilities, KEPCO NF has constructed a new storage facilities for PWR Nuclear Power Plants (NPPs). These facilities are designed to store various types of nuclear materials including uranium hexafluoride (UF_6), uranium dioxide (UO_2) powder, recovered powder, sintered pellets, UO_2 fuel rods, and fuel assemblies. However, as the development of LEU+ nuclear fuel progressed, criticality safety evaluation considering various uranium enrichment became necessary.

This study evaluates the criticality safety of the newly constructed storage facilities for powdered nuclear materials, including UO_2 powder produced in the reprocessing process and recovered powder (U_3O_8 and Grinding sludge) from the sintered pellets manufacturing process. The criticality safety evaluation of the powder storage facilities were conducted using the CSAS6 (Criticality Safety Analysis Sequence) module of the SCALE 6.2.4 Monte Carlo neutron transport code, utilizing continuous cross-section data based on ENDF/B-VII.0.

The criteria for nuclear criticality safety evaluation is that the effective multiplication factor (k_{eff}) must not exceed 0.95 with a 95% confidence level considering all biases and uncertainties under immersion conditions. And under optimal moderation conditions, it must not exceed 0.98. In this study, Criticality safety evaluation was performed according to the above criteria.

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Keywords: Criticality safety, LEU+, Nuclear fuel, Recovered powder, SCALE

Optimizing Uranium Enrichment Analysis: A Comparative Study of ROI and Peak Fitting Techniques in Nuclear Safeguards

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The study aims to investigate the differences and relative advantages of two prevalent techniques used in uranium enrichment analysis which are the traditional two region of Interest (ROI) technique and peak fitting technique. Uranium enrichment analysis is indeed a cornerstone of the nuclear safeguards community, primarily managed and overseen by the International Atomic Energy Agency (IAEA). This process involves determining the isotopic composition of uranium in a given sample, which is crucial for several reasons, notably for monitoring nuclear non-proliferation agreements and ensuring the peaceful use of nuclear materials. Accurate uranium enrichment analysis faces several challenges, including as below. Detection of Low Enriched Uranium (LEU): Differentiating between natural uranium and LEU, which has a slight increase in U-235 concentration, requires highly sensitive and precise instruments. Handling of Highly Enriched Uranium (HEU): HEU poses significant proliferation risks, and its detection and analysis demand stringent security and technical measures. Technical Sophistication: The technical complexity of analysis methods requires specialized equipment, software and highly trained personnel.

The traditional two ROI technique, which integrates counts within predefined spectral regions, offers simplicity and speed but may lack precision in complex spectra. Conversely, peak fitting techniques, which involve modeling the spectrum as a sum of component peaks, promise enhanced accuracy by accommodating spectral overlaps and background variations. This research conducts a comparative analysis of both methods across various sample spectra, focusing on parameters such as accuracy, precision, and robustness to spectral interference. The evaluation employs real-world gamma-ray spectra from enriched uranium samples, providing insights into the suitability of each technique under different analytical circumstances. Findings indicate that while traditional two ROI techniques are advantageous for quick screenings and analyses with minimal spectral complexity, peak fitting techniques excel in scenarios demanding high precision and in the presence of significant spectral overlap. The study contributes to optimizing uranium enrichment analysis, offering guidelines for selecting the appropriate analytical approach based on the specific requirements of nuclear material control and safeguards.

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Keywords: Uranium enrichment analysis, Two region of interest, Peak fitting, Nuclear safeguards

~~A Study on the Awareness of Physical Protection Among Nuclear
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Withdrawal

Enhancement of VR Content for Physical Protection Inspector

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Withdrawal

Highly Selective Cesium Adsorption From Acidic Solution Using Magnetic Calcined ZIF-67/Ammonium Molybdophosphate/Polyaniline Composites

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Due to the recovery of radioactive uranium and plutonium from spent fuel using strong nitric acid, acidic high-level liquid waste (HLLW) contains radioactive cesium as a fission product. Among various fission products, ^{137}Cs must be removed from HLLW because of its long half-life, emission of gamma rays and beta particles, high heat output, and high water solubility. To purify this radionuclide from acidic solutions, magnetic calcined ZIF 67 and ammonium molybdophosphate immobilized in polyaniline (MZAP) were successfully synthesized. The physicochemical structure and stability of MZAP composites were investigated using FT-IR, XRD, VSM, BET, SEM, and TEM instruments. In batch experiments, the adsorption behavior of cesium was studied based on the effects of pH, reaction time, initial cesium concentration, temperature, and the presence of interfering ions. The MZAP composites exhibited excellent adsorption performance with an adsorption distribution coefficient of more than $4.53 \times 10^4 \text{ kg/m}^3$ in the solution pH range of 0 to 11.5. In addition, characterization results revealed that their stability was maintained in 1 M HNO_3 before and after cesium adsorption. The cesium adsorption process followed pseudo-second-order kinetics and the Langmuir isotherm, showing a maximum adsorption capacity of 47.97 mg/g at pH 1 and 25°C with an initial cesium concentration of 500 mg/L. The cesium adsorption mechanism for the MZAP composites was mainly governed by cation exchange between ammonium and cesium ions and electrostatic attraction. Therefore, these findings indicate that the MZAP composites selectively remove cesium from strong acidic solutions, allowing for stable magnetic separation.

Keywords: Cesium, Ammonium molybdophosphate, Polyaniline, ZIF-67, Adsorption, Magnetic separation

Review on Overseas Case of Validity Verification of the Stack Sampling Technical Standard

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Korea is pursuing a continuing operation license for 10 units whose design life is scheduled to be completed by 2030. In order to continue operating, it is necessary to analyze the difference between the latest technical standards and valid technical standards, apply the latest technical standards that reflect operating experience and research results for safety-related SSCs, and conduct safety assessments. The gas radiation monitoring system of nuclear power plants is divided into process radiation monitors and emissions radiation monitors. Samples are collected from the duct through a nozzle and transferred to the monitor through the sample transfer pipe. Among them, the gas emission radiation monitor (or sampler) is installed in the duct at the rear end of the ACU and used to evaluate the air emission of the emissions. The design standard of the sampling system was ANSI/HPS N13.1, which was established in 1969, and then revised to ANSI/HPS N13.1-1999. The nozzle for collecting samples was also changed between the 1969 technical standard and the 1999 technical standard, and there is a significant difference in the technical standard. Therefore, in the case of the 10 units currently in continuous operation that were built before the 2000s and were designed according to the ANSI/HPS N13.1-1969 technical standard, the gas radiation monitoring system was designed according to the ANSI N13.1-1999 technical standard, in which the basic concept of the technical standard was changed. It is necessary to check whether it satisfies. For this purpose, this study reviewed the U.S. case of validity verification of the ANSI/HPS N13.1-1969 technical standard.

Since the U.S. nuclear power plants were primarily designed before 1999, stack sampling systems were designed in accordance with ANSI N13.1-1969. Since then, the standard was revised to ANSI N13.1-1999 (reaffirmed in 2011). Currently, U.S. government facilities related to the production of nuclear materials and waste treatment must meet the ANSI N13.1-1999 design standards. The commercial nuclear industry became the “old model” using the 1969 standard. However, there are cases where the stack and ducts of three nuclear power plants were tested according to the 1999 standard (Columbia Generating Station elevated vent, Westinghouse AP 1000 plant vent and control room ventilation duct). The design criteria for the two versions of ANSI N13.1 are presented, the sampling system is compared, and examples of the stacks and ducts tested according to the 1999 version standard are described.

The purpose of this study was to look at these test results and to examine the applicability to domestic nuclear power plants that are continuing operation.

Keywords: Stack sampling system, Gas emission radiation monitor, ANSI/HPS N13.1, Validity verification

Study of Transportation Risk Assessment Model Input Factor Sensitivity Evaluation Methodology

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The demand for transportation of radioactive waste in Korea is increasing. Prior to the transportation, a risk assessment should be performed. In Korea, for risk assessment, domestic characteristics are considered only for some factors of the overseas assessment model. Accordingly, a transportation risk assessment model reflecting domestic characteristics was developed. If the input factor is sensitive, the result of assessment model fluctuates with the change in the input factor. Therefore, it is necessary to classify the sensitive factor by performing a sensitivity analysis on the developed model. In this study, the sensitivity evaluation methodology was analyzed prior to the sensitivity analysis of the input factor of the assessment model. As evaluation methodologies, 1) sensitivity evaluation through error propagation formula, 2) sensitivity evaluation through latin hypercube sampling (LHS), and 3) sensitivity evaluation through normalized dose difference (NDD) calculation were selected and analyzed the evaluation methods and characteristics. The first method is a sensitivity evaluation through error propagation formula. It is a method of evaluating the sensitivity by partial differentiation of the assessment model for individual input factors and then calculating the amount of change in the evaluation model when the factor changes. It is good to be able to evaluate the sensitivity of each detailed section of individual factors. The second method is the sensitivity evaluation through LHS. In LHS, a random representative value is selected for each range by evenly dividing the probability distribution of factors. It is a method of generating a factor combination through multi-variable random sampling in which each representative value is randomly matched for each factor. By calculating the dose through the generated factor combination, the dose average and the width of the change according to the factor change can be known. The advantage is that the validity of the selected input factor can be verified through the average and standard deviation of the results obtained through sampling. The final method is a sensitivity evaluation through NDD calculation. One of the probabilistic sensitivity evaluations is a method of evaluating the sensitivity of the evaluation result to the input factor through changes in the dose obtained through the high, low values and default values of the factor. It is good to analyze the overall sensitivity of individual input factors to the result with only information about high, low, and default values. The result of this study can be used as the basis for analyzing the input factors sensitivity of the transportation risk assessment model developed according to domestic characteristics.

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Keywords: Overland transport, Radioactive waste, Sensitivity analysis methodology

Activation Analysis of ISOL Target Coolant in RAON

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The ISOL (Isotope Separation On-Line) has the important role of creating and separating the rare isotopes in the RAON (Rare isotope Accelerator complex for On-line experiments). It is composed of Target Ion Source Module Station (TIS Module Station) and RIB Separation System. Basically, the 70 MeV cyclotron provides the proton beam to the SiC target. The rare isotopes generated from the SiC target are separated as passing through the RIB Separation System. All of these process involve the activation. As the operation of the ISOL gets powered up, the concern about the activation becomes more important. The activation in the radiation facility is divided into two parts, the air activation and the coolant activation. In case of the TIS module station, the vacuum exhaust system had built separately and has been under special management. The TIS target coolant circulates only in the TIS Module Station and it's not emitted in principle, which secure the radiation safety of the ISOL facility and no fatal risks were detected. However, we have collected the coolant samples and measure the activity periodically.

In this paper, we focused on the coolant activation and we measured the ^3H and ^{14}C of the TIS target coolant using the Liquid Scintillation Counter (LSC). Before measuring the activation of the TIS target coolant, we calculated the concentrations of the radionuclides using the activation analysis equation, which could be compared with our measurement results.

The result shows that the activities of ^3H and ^{14}C are minor levels compare to the emission control standards, because the experiment of the ISOL facility is in the beginning. So we expect that further measurement will be based on this results. Consequently, this paper was prepared for the purpose of obtaining the basic standards of the activation analysis in the ISOL target coolant.

Acknowledgements

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Keywords: RAON, ISOL, TIS, LSC

7분과

방사화학 (ORAL)



Synthesis of Uranium Chloride Through Hydriding and Chlorination of Uranium Metal for the Fuel Production of Molten Salt Reactor

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The allure of fast spectrum molten salt reactors is on the rise, driven by their potential for high power density and their knack for effectively managing high-level nuclear waste. Among the myriad fuel options for FS-MSRs, chloride-based fuels stand out for their superior actinide solubility compared to fluoride-based salts. Consequently, chloride-based FS-MSRs are gaining traction as a solution for handling spent nuclear fuel.

In this arena, high-assay low-enriched uranium and U/TRU metal fuel sourced from pyroprocessing technology emerge as the most viable candidates for chloride-based MSR fuel. Yet, the journey from metal to chloride form presents its own set of challenges. Conventional methods involving chlorinating agents like CdCl_2 and BiCl_2 often yield unwanted by-products and sluggish reaction rates. Similarly, chlorination with chlorine gas poses toxicity concerns and generates undesirable compounds.

In a previous study, NH_4Cl was employed to chlorinate U metal into UCl_3 within an airtight reactor. While this process successfully chlorinated the U metal into chloride forms, undesirable impurities resulting from the reaction between NH_4Cl and the reactor wall were detected in the final product. This emphasized the need for further research and optimization in the chlorination process to obtain high-purity MSR fuel. Therefore, this study focuses on redesigning a chlorination reactor to eliminate any interaction with the reactor wall. This goal was accomplished by incorporating an Al_2O_3 crucible within the reactor. Additionally, when we expanded the reactor dimensions to achieve UCl_3 production exceeding 100 g, we observed incomplete chlorination of U pellets. To address this, prior to chlorination, U metal chunks were transformed into UH_3 powder by reacting them with H_2 gas inside the reactor. The resulting UH_3 particles had a size of less than 5 μm . Considering that UH_3 decomposes to U metal at temperatures above 673 K, we directly utilized the UH_3 powder to produce pure UCl_3 . This production process involved reacting UH_3 powder with NH_4Cl at a temperature of 753 K. The purity of the UCl_3 product was confirmed through different analyses. Experimental findings will be presented and deliberated for further discussion.

Acknowledgements

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Keywords: Molten salt reactor, U tri-chloride, Hydriding, Chlorination, Pyroprocessing

Corrosion Behavior of Structural Materials for Molten Salt Reactor in NaCl-MgCl₂ Salt

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Molten salt reactor (MSR) uses fluoride or chloride based molten salt as a coolant of the system, and fuel materials are dissolved in the molten salt, therefore it can be act as both coolant and nuclear fuel. A few issues have arisen from early-stage research and development program of MSR from Oak Ridge National Laboratory, including corrosion of structural materials and fission product management.

And it is essential to build a loop system for basic corrosion and fission product management experiment. In this study, the design criteria of high-temperature molten salt natural convection loop is addressed, and based on the criteria, high-temperature molten salt natural convection loop for chloride based salt is designed and manufactured. Before injecting NaCl-MgCl₂ salt into the loop, step-by-step dehydration method is applied to remove moisture in the chloride salt. Weight difference between pre-treatment and post-treatment salt is investigated to clarify the amount of moisture in the pre-treatment salt. Prepared chloride salt is injected into the fill tank of the natural convection loop, as well as corrosion coupons for candidate structural materials for molten chloride salt reactor, SS304, SS316, and Hastelloy-N, are also prepared. Hot leg temperature of molten salt natural convection loop is set as 650°C, while cold leg temperature of molten salt natural convection loop is set as 550°C to generate flow in the natural convection loop. Each heater installed in the molten salt natural convection loop has its own current sensor to calculate the molten salt flow rate by heating power to hot leg and cold leg of molten salt natural convection loop. During the experiment, a vacuum pump connected to fill pot is operated to maintain degree of vacuum under 5 mTorr. After 30 d corrosion experiment, the salt is drained from the loop to the drain tank for inductively coupled plasma optical emission spectroscopy (ICP-OES), to investigate metal dissolution from structural materials to salt. Corroded candidate structural materials are analyzed with scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDS).

For investigating the effect of additives on corrosion of structural materials, Mg(OH)₂ and MgCl₂*6H₂O are added into the NaCl-MgCl₂ eutectic salt. Prepared chloride salt is injected into the autoclave in the glove box, as well as corrosion coupons for candidate structural materials for molten chloride salt reactor, SS316, Alloy 600, and C-276 are also prepared. The temperature is set as 700°C. After 500 h corrosion experiment, the samples are taken out from the autoclave, and they are analyzed with SEM and EDS. SS316 samples show weight loss with all salt conditions, while Alloy 600 and C-276 show weight gain after the corrosion experiment.

Keywords: Molten salt reactor, NaCl-MgCl₂, Corrosion, Stainless steel, Ni-alloy

Unary Molten Salt Synthesis of Tunable Adsorbents and Photocatalysts for Uranium Extraction From Seawater

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Extracting uranium from seawater would beneficially contribute to the sustainable utilization of nuclear energy. A green, efficient, and economical strategy has long been studied but still faces challenges of low uranyl concentration, high salinity, and competition with carbonate ions. Conventional adsorbents are regarded as an efficient way to extract uranium from seawater with the benefits of low cost, simple process, and high feasibility. In addition, hybrid approaches combining adsorption with electrochemistry or photocatalysis have proved to be a potential solution for uranium extraction from seawater too. Molten salt synthesis is one of the common methods used for the synthesizing and functionalization of photocatalysts including carbon nitrides and is beneficial in improving their photocatalytic activity. Here we synthesized a series of carbon nitride materials in different unary molten salts, and reported different tunable carbon nitrides whose adsorption capacity and photocatalytic activity can be adjusted with the synthesis temperature, raw materials ratio and molten salt system. The carbon nitride nanostructure obtained at 550°C (CN550) exhibited the highest photocatalysis-assisted extraction ability (9.4 times higher than that of g-C₃N₄), uranyl ions could be captured on the surface of CN550, and then deposited as metastudtite nanoparticles with light illumination. Compared with the pure physicochemical adsorption method, a tenfold higher uranium extraction capacity of up to 1,556 mg g⁻¹ was achieved using the photoactive CN550. Further it was found to work under natural sunlight illumination as well. Similarly, carbon nitride obtained at 600°C (CN600) showed the highest adsorption capacity of 1,387.0 mg g⁻¹, which is 51 times higher than that of g-C₃N₄. In addition to the outstanding adsorption ability, CN600 has a verified extraction rate of nearly 100% of uranyl carbonate from simulated seawater with a pH of 6.0, 7.1, or 8.0.

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Keywords: Carbon nitride, Uranium extraction, Seawater, Adsorption, Photocatalysis, Molten salt

Note: The present work constitutes work recently published by the authors.

Range of Coefficients of Variation for Elemental Concentrations in Volatile Aerosols Generated From Molten Iron-Concrete Reaction

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Due to their ability to facilitate the mobility of fission products, radioactive aerosols have garnered significant attention from researchers. In the event of a severe nuclear power plant accident, aerosols are generated from the interaction between core melt and basemat concrete, potentially facilitating the diffusion of radioactivity. To investigate this phenomenon, we conducted experiments involving the reaction of molten iron and concrete in the presence of simulated nuclear fission products, subsequently measuring the elemental concentrations of the resulting aerosols.

We expressed the element concentrations of the simulated fission products relative to the concentrations of the basic aerosol-forming elements (Si or Fe). These relative concentrations serve as indicators of element volatilities during the reaction. Additionally, we utilized the coefficient of variation (CV) of the relative concentrations to assess the reliability of our aerosol generation tests. Our results revealed CV values of less than 20% for basic aerosol elements, less than 30% for easily volatilized elements, and over 100% for elements with low volatility. These findings provide valuable insights for evaluating the reliability of aerosol generation tests.

Acknowledgements

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Keywords: Coefficients of variation, Aerosol, Molten iron, Concrete, Fission products

Determination of Radioactive Concentrations for Internal Structure of Reactor Vessel in Kori Nuclear Power Plant Unit 1

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This study presents gamma radioactive concentrations for baffle-former bolt and surveillance specimen sampling from reactor vessel in Kori nuclear power plant (Unit 1). The metal samples taken from a baffle-former bolt and surveillance specimen were dissolved in concentrated HNO_3 1 mL and concentrated HCl 3 mL (aqua regia) on hot plate. The stock solution was made from dissolved samples with distilled water weighing accurately the mass with a scale with 4 decimal places. A small amount of the sample taken from the stock solution was put in 20 mL plastic vial. The sample in plastic vial was put on plastic holder at 10 cm distance between sample and detector. The sample was measured the gamma radioactive concentration by a HPGe gamma spectrometer after calibration with CRM (certified reference material, Eckert&Ziegler). The measured gamma radioactive concentrations in baffle-former bolt and surveillance specimen were validated with calculated values by computer simulation using ACT2 computer program.

Keywords: Gamma radioactive concentration, Baffle-former bolt, Surveillance specimen, HPGe gamma spectrometer, ACT2 computer program

Consideration of the Standard (21 CFR 11) for Electronic Records and Electronic Signature in Radioactivity Analysis

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Radioactivity analysis results are mostly used as the basis for other works, such as the basis for the regulation of radioactive waste, the basis for safety, and the basis for determining whether to perform other work, so the objectivity and reliability of the radioactivity analysis procedure are more important. Reliable analysis results require data to be verified throughout the entire process from sample collection to report writing. Currently, most radioactivity analysis interprets measurement signals and calculates results using programs and algorithms built into each analysis device, so if the analysis device has been properly calibrated, the possibility of error in the analysis device itself is extremely low. Although history and data management are performed in some laboratory management programs such as LIMS, the programs are written according to the experience and intention of each program developer, and there are no clear standards for electronic records and electronic signatures.

In this study, as one of the ways to ensure the reliability of radioactivity analysis results, the need to introduce 21 CFR 11, the final rule on electronic records and electronic signatures for all operations regulated by the U.S. FDA, to the domestic radioactivity analysis industry was considered. 21 CFR 11 was announced in March 1997 and went into effect on August 20, 1997. Its goal is to ensure that electronic records and electronic signature systems are reliable, and it is a standard that is already being used in the bioanalysis field. 21 CFR 11 consists of three sections: General Provisions, Electronic Records, and Electronic Signature, and the electronic records and electronic signature sections present various conditions to comply with 21 CFR 11. The electronic records section stipulates system verification, detection of altered or invalid records, audit trail, training, record retention, data reproduction, information at the time of signature (name, reason, date, time), removal or copying, etc. The electronic signature section specifies technical requirements and procedural and administrative control requirements related to electronic signatures, such as identity verification of the electronic signer, consistency between electronic signatures and handwritten signatures, controls to prevent fraud, and detection of unauthorized access.

Since most radioactivity analysis results are the basis for other works, if an electronic record and electronic signature system that satisfies 21 CFR 11 is technically included in the radioactivity analysis system, it will fundamentally prevent human errors and improve the reliability of the analysis results.

Keywords: 21 CFR 11, Electronic records, Electronic signature, Radioactivity analysis

Determination of ^{129}I Iodine in Cement Waste Form

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The Korea Atomic Energy Research Institute (KAERI) plans to transport the cement waste to a final disposal facility after it meets acceptable standards for leaching, compressive strength, and radionuclide inventory. In order to satisfy the suitability for disposal, it is particularly important to accurately analyze the concentration of ^{129}I , a radionuclide present in the contaminated concrete. In addition, ^{129}I is a radioactive element produced when uranium is fissioned in a nuclear reactor and has a long half-life ($t_{1/2}=1.57\times 10^7$ y), can be absorbed and accumulated in the body, and releases less energy than cesium or cobalt, making it difficult to assess nuclide stocks.

In this study, we focused on the determination of iodine in the form of cement waste using KI for iodine recovery. For particle size dependent evaluation, sieving of cement waste was performed and ^{129}I (11.083 ± 1.019 Bq/g) was added to 1 g of cement waste particles for each size. As a result, we observed that the iodine yield decreased with increasing cement waste particle size (0.840, 0.582 at $\phi > 6.35$ mm, $\phi < 0.5$ mm). To improve the ^{129}I yield from cement waste particles, we added a reduction step in the process of extracting ^{129}I from cement waste. After adding the reduction step, the iodine yield achieved more than 90% for all sizes of cement waste particles. In addition, our experimental results also showed removal efficiencies of more than 99% for ^{137}Cs and ^{60}Co .

In conclusion, the high recovery of ^{129}I in cement by size with the addition of the reduction step is expected to enable quantitative analysis of cement in large quantities of decommissioning waste without the need for particle size segmentation.

Acknowledgements

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Keywords: Iodine, Cement waste, Nuclear decommissioning waste, Acid leaching

7분과

방사화학 (POSTER)



Separation of Iron and Nickel in Radioactive Waste Sample by Cation Ion Exchange Chromatography

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The radioactivity concentration of radionuclides must be determined in order to permanently dispose radioactive waste drums. To measure the radioactivity concentration of non-volatile beta emitting nuclides, it must be separated and then measured because of interference from coexisting nuclides. Sequential chemical separation procedure of non-volatile beta emitting nuclides has been using to measure radioactivity concentration of ⁹⁹Tc, ⁹⁰Sr, ⁵⁵Fe, ⁹⁴Nb, ⁵⁹Ni and ⁶³Ni in radioactive waste samples. Technetium and strontium were separated by anion exchange chromatography and co-precipitation, respectively. Most of the nickel, iron and niobium were present in the supernatant from which the co-precipitate [Ca(Sr)-C₂O₄] was separated. Nickel was passed through the anion exchange resin column in 0.1 M NH₄-C₂O₄ (pH 4.5–5), while iron and niobium were adsorbed on anion exchange resins. Because iron binds to the anion exchange resin, the maximum iron content that can be separated by this procedure is 50 mg. According to a literature search, it was found that the distribution coefficient of nickel was greater than that of iron in cation exchange resin and 0.2 M HCl/more than 80% of acetone. Therefore, cation exchange chromatography was developed to separate iron and nickel from radioactive waste containing more than 50 mg of iron. Up to 300 mg of iron could be separated from nickel using a mixed acid (0.2 M HCl/95% acetone) as an eluent in the cation exchange resin column. Nickel adsorbed on cation exchange resin was desorbed with 10 M HCl.

Keywords: Iron, Nickel, Separation, Cation exchange chromatography, Radioactive waste

Properties of Uranium Nanoparticles for Solubility Test in Anoxic Groundwater

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The radionuclide (RN) migration processes and the dissolution behaviors of relevant solubility-limiting solid phases (SLSPs) of RNs should be well understood to properly evaluate the long-term safety of a deep geological repository (DGR) system. Recently, colloidal forms of RNs, e.g., Pu from nuclear testing sites, were found as major species responsible for subsurface migration along natural groundwater systems. Moreover, the solubility of RN colloids, such as $\text{ThO}_2(\text{coll})$, were reported to vary depending on their aging history, possibly owing to the changes in crystallinity of the inner particle structure as well as their sizes. In our previous study of geochemical modeling of uranium (U) solubility, $\text{UO}_2(\text{s})$ or solid forms of U(IV) was shown to be the SLSP of U in a reducing groundwater system. Also, we reported that U(IV) nanoparticles (NPs) or their pseudo-stable colloids can be prepared, which exhibit $\text{UO}_2(\text{cr})$ -, or uraninite-crystalline structures in their particle core. In this study, we attempted to obtain three different types of NPs: one with loosely-connected (lc) structures of primary particles (PPs), another with a standard (std) structure of aggregated PPs, and the third with further aggregated (aged) structures of U(IV)-NPs. Various analytical techniques were utilized for the NP characterization including dynamic light scattering (DLS) analysis, zeta potential measurement, nanoparticle tracking analysis, transmission electron microscopy (TEM), and UV-Vis absorption spectroscopy. Typically, NPs (std) exhibit ~ 30 nm of an average particle size and $> +40$ mV of a zeta potential value. The aging of NPs over time gradually induces de-stabilization of aqueous NP colloids, in that, the solutions of NPs (aged) show lower values of zeta potential ($< +30$ mV) with increased particle sizes due to further aggregation ($> 1 \mu\text{m}$). Interestingly, for NPs (lc) the results of DLS and zeta potential measurements did not show significant difference from those of NPs(std). However, TEM image analysis was found useful to clearly demonstrate the difference between NPs (lc) and NPs (std) in their morphological structures and the size of crystalline PPs, varying from ~ 2 nm up to ~ 8 nm. Thus, the apparent surface area of NPs is expected to be in decreasing order, i.e., $\text{NPs(lc)} > \text{NPs(std)} > \text{NPs(aged)}$. These three types of U(IV)-NPs are candidates of U(IV) SLSP for measuring solubility using simulated groundwater for the DB3 granitic groundwater composition of KURT. The expected outcome of the solubility tests and the further characterization methods for these U(IV)-NPs will be discussed in the presentation.

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Keywords: Tetravalent uranium, Solubility, Nanoparticles, Colloidal stability, Zeta potential

Investigation and Analysis of Cs-137/I-129 Nuclide Concentrations and MDA Data Characteristics for I-129 Nuclide in Radioactive Waste

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To determine the concentration of I-129, an evaluation method using correlation with a key nuclide such as Cs-137 is usually used. In operating nuclear power plants, I-129 nuclides are present in very low concentrations in radioactive waste, with a significant portion below the detection limit. In order to determine the nuclide inventory, the concentration of the DTM nuclide I-129 in radioactive waste is calculated from the concentration of Cs-137, so the concentration of I-129 is likely to be overestimated even if it is below the minimum detectable activity (MDA).

This study was conducted for the purpose of investigating and analyzing the possibility of overestimation that occurs when applying the evaluation of the radioactive waste in radioactive waste (hereinafter, “decommissioned waste”) resulting from the decommissioning of a nuclear power plant. Once the MDA of I-129 and the concentration range of major nuclides were confirmed, we attempted to present a methodology that could predict the possibility of overestimation when applying the evaluation method.

For test simulation, from the nuclide concentration data of the major nuclides Cs-137 and I-129 based on the field experience of the previous researcher, the concentration data of four Cs-137 and I-129, including the data of MDA (“CASE 1 concentration data”) was selected. Based on CASE 1 concentration data, random numbers were generated using a Python program to generate random concentration distribution data (50 items) of Cs-137 in the concentration range of 0.15 to 150 Bq/g, and the corresponding I-129 Concentration data was derived. Additionally, a linear regression model and model parameters were obtained from the data distribution. By applying the model, we investigated the proportion of I-129 concentrations below or above MDA and the dependence of other Cs-137 concentrations.

For another test simulation, 45 Cs-137 and I-129 concentration data (CASE 2 concentration data) were used, including waste resin, DAW, CVCS filter, activated carbon, and soil, which are samples during operation of a low-level commercial nuclear reactor.

As a result, tests using CASE 1 concentration data showed that the concentration of I-129 in decommissioning waste could be overestimated by up to 98%. In the case of tests using CASE 2 concentration data, I-129 concentration showed the possibility of overestimation by up to approximately 47%.

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Keywords: Decommissioning waste, I-129, Radionuclide inventory, DTM, Nuclide analysis, Key nuclides

The Effect Evaluation Following as Amine Saturation Operation for Reducing Corrosion Products of Component Materials in Secondary System at NPPs

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The main material of the secondary system at NPP is composed of carbon steel, and is occurred flow-accelerated corrosion (FAC) and general corrosion during normal operation. The generated corrosion products are flowed into the steam generator through the feedwater and accumulated. The accumulated corrosion products provide an environment where impurities can be concentrated. This can also be a major cause of corrosion (Stress Corrosion Cracking, Denting, Flow Plugging, etc.) in the steam generator components. To minimize corrosion in the secondary system's piping and equipment, one can consider changing to corrosion-resistant materials or increasing the pH of the system. In order to reduce corrosion, NPPs changed to low-alloy steel as a corrosion-resistant material. Additionally, KHNP CRI was developing amine saturation operation technology for condensate polishing plant (CPP) to increase pH.

In this study, the amine saturation operation of the condensate polishing plant (CPP) was used to evaluate the effect of reducing corrosion in the secondary system through field application in KSNP (Korea Standard Nuclear Power Plant). To evaluate the corrosion reduction effect, corrosion product sampler (CPS) were installed at three locations; the feedwater system, condensate pump outlet, high-pressure feedwater heater. Iron concentration was analyzed once a week. As a result of amine saturation of CPP, the pH of the feedwater system increased from 9.80 to 9.87, and the pH of the condensate pump outlet and high-pressure feedwater heater also increased. And the amount of iron oxide was reduced at each major point according to the amine saturation operation, especially iron oxide was reduced by approximately 45% in feedwater system. The corrosion products at condensate pump outlet and high-pressure feedwater heater decreased by 10% and 18%, respectively. Due to decreasing corrosion products, the amount of deposited in the steam generator was reduced by approximately 49%.

In these results, the pH increases through amine saturation operation. As the pH increased, the corrosion of components material in secondary system was reduced. As a result, the amount of the corrosion products accumulated in the steam generator decreased. Therefore, it is judged to be helpful in maintaining the integrity of the steam generator tubes.

Keywords: Amine saturation operation, pH, Corrosion, Iron oxide, Secondary system

Study on Sample Pre-Treatment Methodology for Nuclide Analysis of Decommissioning Concrete Waste

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Unlike waste generated during the operation of a nuclear power plant (NPP), NPP decommissioning waste is generated in large quantities over a short period of time and consists of more than 70% concrete and soil. In order to dispose of radioactive waste, it is necessary to determine the concentration of 14 nuclides in accordance with the disposal regulation. To determine the concentration of nuclides, pre-treatment and chemical separation of samples must be carried out. In this study, microwave acid digestion (MAD) and acid leaching (AL) methods for nuclide analysis of concrete waste were tested and the results were discussed.

A simulated sample of concrete powder (particle size < 300 μm) with 9 elemental standards (Ce, Co, Cs, Fe, Nb, Ni, Re, Sr and U) were prepared into a solution using the MAD and AL method, respectively. MAD used 1.00 g of concrete powder and AL used 10.0 g of concrete powder. Upon completion of pre-treatment using both methods, the chemical yield of 9 elements were determined using ICP-OES and ICP-MS. The findings revealed that all elements were recovered quantitatively, exceeding a threshold of 92%. However, MAD has the advantage of being a very efficient method for complete digestion of small quantities, requiring less sample volume of acid solution and producing less secondary waste, while AL has the advantage of being fast and simple, especially when large enough sample quantities are required to lower minimum detectable activity (MDA). MAD and AL have been experimentally confirmed to be suitable for sample preparation for concrete waste analysis. Since each method has its own advantages and disadvantages, it is recommended to utilize the appropriate method according to the waste disposal strategy based on the radionuclide concentration limit for low and intermediate-level radioactive waste or clearance.

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Keywords: Decommissioning, Radiochemistry, Concrete, Radionuclides analysis, Pre-treatment

A Study on Sample Drying Conditions for Soil Waste Gamma Spectroscopy

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Various radiometric methods are employed for the permanent disposal of low- and intermediate-level radioactive waste in Korea. Among those, gamma spectrometry is the most commonly utilized non-destructive method for analyzing gamma nuclides present in wastes. Moisture content in soil samples can significantly influence the accuracy of radioactivity analysis when examining gamma nuclides. Therefore, it is imperative to dry the samples before conducting measurements.

In this study, the optimal drying time for analyzing soil samples at 110°C were investigated using the gravimetric method. Specifically, we found that 8 hr-duration of drying was sufficient to ensure complete dryness of the sample.

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Keywords: Radioactive waste, Radionuclide inventory, Low- and intermediate-level radioactive waste, Gamma-ray spectroscopy

Preliminary Study on Monte-Carlo Simulation of Gamma Detector for Low-Level Solid Radioactive Waste

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The main analytical method used to evaluate the radioactivity of gamma nuclides for permanent disposal of radioactive waste is gamma spectroscopy. However, in solid samples such as soil and concrete, there is a density difference between the calibration sample and the analyte. One of the ways to compensate for this density difference is to calibrate the instrument using a calibration sample with exactly the same characteristics as the analyte. However, since it is physically impossible to prepare a calibration sample with exactly the same characteristics for every analyte. Therefore, a radiation simulation program is usually used to simulate the same measurement environment as in the laboratory, and then the detector's measurement efficiency is calculated from the simulation results to derive the radioactivity value of the sample. KAERI Radioactive Waste Chemical Analysis Center uses a simulation program called MCNP, Monte Carlo N-Particle code to calculate the measurement efficiency of gamma detectors. In this study, we summarize various tally parameters that should be considered when calculating the detector's measurement efficiency with Pulse Height Tally of MCNP and report on the effect of using these parameters on the simulation results.

Acknowledgements

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Keywords: HPGe, Monte-carlo simulation, MCNP, Coincidence summing, Radioactive waste, Concrete, Gamma spectroscopy

The Safety Assessment About Generation of Radioactive Waste for Zinc Injection in Primary Coolant at NPPs

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Zinc injection is a technology that stabilizes the metal oxide film structure by injecting zinc into Reactor Coolant System (RCS) for the purpose of reducing primary system radiation dose and improving material corrosion resistance. Zinc is injected into RCS through Chemical and Volume Control System (CVCS) by dissolving depleted zinc acetate dihydrate in demineralized water. In domestic nuclear power plants, zinc aqueous solution is injected into RCS at a concentration of 5 ppb for the purpose of reducing radiation dose.

In this study, the effect of zinc injection on radioactive waste generation in nuclear power plants was evaluated. Zinc injection into the RCS does not affect gaseous radioactivity concentration, therefore emissions to gaseous radioactive waste disposal systems do not increase. On the other hand, liquid radioactive waste, the system capacity was increased following as the zinc aqueous solution is injected, but the amount of zinc aqueous solution injected is negligibly small compared to the makeup water injected for reactivity control, and since it is recovered by the boric acid water recovery system, it does not affect the change of liquid waste.

In the case of solid radioactive waste, corrosion products are expected to temporarily increase due to zinc injection, so it is necessary to evaluate the impact on the ion exchanger and the purification filter. Even if the load on the ion exchanger increases due to temporary increase in corrosion products, CVCS purification resin can maintain sufficient ion exchange capacity, and in the case of domestic nuclear power plants, CVCS resin is replaced every overhaul after one cycle of use, so there is no additional waste generation. In the case of CVCS let down purification filters, a temporary increase in replacement amount is expected in the early stages of zinc injection application due to an increase in corrosion products. But as a result of comparing the filter replacement history before and after zinc injection in power plants where applied zinc injection for several cycles, no clear increase or decrease trend was found. Therefore, it was confirmed that it did not have a significant impact on the increase in solid radioactive waste.

Keywords: Zinc injection, Radioactive waste, Primary system, Corrosion

The Consideration of Injecting Method for the First Zinc Application at AOA (Axial Offset Anomaly) Experienced NPPs

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Zinc injection operation is purpose to stabilize the metal oxide film structure to reduce primary system radiation dose and improve material corrosion resistance. The main materials of the primary system are stainless steel and nickel alloys, and Ni and Fe are the main chemical composition. When zinc is injected into the primary system through the zinc injection operation, Ni and Fe are detected as representative metal ions, and Ni in particular is used as the main diagnostic variables.

In the NPP, which did not experience AOA, by injecting 1,428 ppm of Zn aqueous solution (This concentration is calculated value for adjusting the concentration of Zn in RCS to 5 ppb), there were no cases of rapid increase in Ni Con. in RCS. However, in the purpose in this paper, the NPP that experienced AOA before applying Zn injection, and after 18 days of Zn injection, the Ni Con. exceeded the water limit. Therefore, the Zn injection was stopped, and after the Ni Con. Stabilized within the water limit, the Zn solution diluted to 550 ppm was injected. Since then, the Ni Con. Has remained stable within the water limit, and the Con. Of other metal ions has also been maintained without problems.

There are factors to consider before conducting Zn injection, such as daily injection concentration and other relevant considerations. Based on the experience of KHNP, when initially Zn injection operation, gradually increasing the injection Con. from a lower (less than Zn 5 ppb in RCS) level seems to minimize the physicochemical impact within the water chemistry system of the nuclear plant that experienced AOA, compared to starting with a high injection con. Considering the above experience, it appears reasonable to consider the operational approach for PWR nuclear power plants that will apply zinc injection in the future.

Keywords: Initially zn injection operation, Zinc concentration, AOA

The Effective Separation Method and Validation of Transuranic Elements (TRU) Such as Pu, Am, and Cm

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The nuclear-related facilities produce the radioactive wastes of large amounts every year, and the process of determining the concentration of radionuclides is essential to deal with it. In this study, transuranium elements (TRUs) were effectively isolated and purified using ion exchange resins. TRUs (Pu, Am and Cm) generated from the fission products are separated through an appropriated column process and quantified by alpha spectroscopy (^{238}Pu , $^{239+240}\text{Pu}$, ^{241}Am , ^{242}Cm , ^{244}Cm), ICP-MS ($^{239}\text{Pu}/^{240}\text{Pu}$) and LSC (^{241}Pu). Finally, it was quantified by effectively isolating and purifying Pu, Am and Cm from the radioactive wastes. And it is evaluated the effectiveness for the separation and quantification of TRUs.

Acknowledgements

The periodic verification (5th) service for the scale factor (SF) of radioactive waste.

Keywords: Transuranic elements (TRU), Pu, Am, Cm

Accelerating Development of Structural Materials for Molten Salt Reactors: A Fast Screening Approach for Evaluating Corrosion Resistance

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A Molten Salt Reactor (MSR) system, one of the Generation IV nuclear reactor types, is a system that utilizes molten salt as a fuel and/or coolant. Since the first MSR research initiated in the mid-1950s, the selection of suitable structural materials has always been a challenge that hinders the timely commercialization of MSRs. The structural materials in a MSR are exposed not only to the high temperatures (over 600°C) and high level of radiation, but also to the highly corrosiveness nature of the molten salt itself. Therefore, structural materials in a MSR must exhibit exceptional resistance to corrosion attacks from the molten salt.

However, current corrosion experiments take several hundreds of hours to test a single specimen, and the experimental preparation methods and environments differ, making it difficult to compare the results quantitatively and reliably. As a result, a huge delay in obtaining results from molten salt corrosion experiments is present. To accelerate the development of the structural materials for MSR, a systematic corrosion test method is required.

Therefore, in this study, an endeavor to develop a fast screening method for verifying the feasibility of alloys as MSR structural materials is proposed. Thermal pretreatment process was conducted in order to purify the metal impurities and moisture in as-received NaCl and MgCl₂ salt. Purified salts were fabricated to pellets of the same size, and SS316 alloy specimens were polished to keep the salt contact area constant. Salt pellets were then placed above the alloy specimens and subsequently heated up to 600°C in the glovebox filled with argon gas. The degradation of the surface as well as the corrosion depth inside the alloy specimens were investigated by scanning electron microscope coupled with energy dispersive X-ray spectroscopy. Results indicate that the moisture and metal impurities contained in the salt attribute to the acceleration of corrosion of the alloys. The degree of polishing of the alloys was also a variable that affects the corrosion behavior, by bringing a change in the salt contact area, thus highlighting the importance of proper and consistent specimen preparation procedures.

In conclusion, an approach for fast screening the corrosion resistance of alloys in contact with molten salts is proposed. This developed method is powerful as it could be applied to any types of alloys, and dozens of specimens could be tested in a single experiment by stacking layers of containers in the furnace.

Acknowledgements

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Keywords: Molten salt reactor, Corrosion, Material development, Fast screening

Development of Rapid Analysis Method for Chelates Contained in Radioactive Waste

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Chelating agents have been used as detergents for the removal of various metal oxides in nuclear power plants, resulting in the generation of cleaning effluents.

According to Article 16, Paragraph 2 of the Nuclear Safety and Security Commission Notice No. 2020-11 of the Intermediate and Low Intermediate Level Radioactive Waste Delivery Regulations, wastes containing chelating agents must have their properties removed during the treatment process, or wastes containing more than 0.1% must be specified, and must be limited to comply with disposal facility requirements.

Therefore the chelating agents present in the waste should be analyzed and quantified before disposing of radioactive waste.

The chelating agents commonly include EDTA, NTA, and citric acid. There are various methods for analyzing chelating agents, and one of the representative techniques is UV-VIS spectroscopy. However, the UV-VIS method has the difficulties of requiring a long experiment time and handling a variety of chemicals, and also has the potential to cause human error depending on the operator's skill level due to complicated experiment conditions.

To improve these shortcomings, we have developed a faster and more accurate Sequential One-Shot Analysis (SOSA) method based on Ion Chromatography (IC) technology. This method has the following advantages:

The SOSA allows simultaneous analysis of multiple chelating compounds. We verified this capability by analyzing representative chelating agents such as EDTA, NTA, and citric acid. The conventional method was required different chemical treatments for each chelating agent, but SOSA enables analysis of three chelating agents without separate pretreatment, securing results within 60 minutes.

The previous method was sensitive to changes due to environmental conditions (temperature, humidity, etc.) and reagents, the SOSA simplified the analysis conditions and minimized the influence of external conditions, thereby increasing data reliability and reproducibility, and minimizing human errors.

Finally, the SOSA is shown lower LOD and LOQ by reducing the number of reagents used compared to previous methods. It allows analysis of trace samples, enhancing its practicality.

Based on these results, SOSA is expected to have an effective impact on chelating agent analysis methods by increasing analyst exposure stability, reducing human error, and improving data reliability.

Keywords: Chelate, Chelating agents, Radioactive wastes, EDTA, Ion chromatography

Effect of Oxygen and Moisture on Fission Product Release Behavior in the Operation of KCl-UCl₃ Molten Salt Reactor (MSR)

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Moisture and Oxygen are the major impurities affecting MSR performance by reducing fuel-salt lifetime and accelerating corrosion in structural material. For long-term operation of MSR, such impurities should be controlled by purity management system and the design of the system requires the chemical form and amount of various fission products including oxides and hydroxides induced by impurities. In this study, for the derivation of such information, Gibbs energy minimization approach using FactSage was applied with extended MSTDB-TC database including the model parameter of two liquid quasi-chemical model for 4 binary system such as ZrCl₄-KCl, PrCl₃-KCl, YCl₃-KCl, SmCl₃-KCl and the thermophysical properties for PmCl₃. For KCl-UCl₃ based MSR with thermal power less than 10 Mw and operating life time of 20 year, total inventory of all fission product in element were firstly calculated using OpenMC code allowing depletion during neutronic calculation. Then, among all inventory, 46 element species from Chloride to Americium were given to the input for equilibrium module of FactSage with its mass for the identification of chemical form of fission products. Moisture and Oxygen was assumed to preferentially exist in base salt and also be continuously ingressed into salt during reactor operation. It was found that in the early stage of reactor operation, major gas species was found to Xe, HCl, Kr, HI, Ar, SnH₄, Cd, however, Xe, Kr, Ar, Cd, InI, CsI in the final stage of operation. In solid and liquid phases, uranium oxychloride, UOCl, of the solid phase and UCl₄ were found to be formed due to the impurity.

Acknowledgements

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Keywords: Fission product, Molten salt reactor, Chemical form, Gibbs energy minimization

Optimization of Pre-Treatment Conditions for Metal Waste Samples at KAERI

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Accurate analysis of radionuclides is important for the permanent disposal of radioactive wastes. Nuclear Safety and Security Commission specified 14 nuclides that we have to identify the nuclide-specific concentration levels, and we follow the notices to dispose the radioactive wastes. Sample pre-treatment, the first step of analysis process, is essential because the radionuclides contained in the sample should be well dissolved so that the concentrations of each nuclide can be analyzed precisely. Pre-treatment must be done differently depending on the physical and chemical properties of the sample.

In this study, we investigated suitable conditions for melting metal waste samples completely. We used a solution comprised of hydrochloric acid, nitric acid, hydrofluoric acid and hydrogen peroxide in appropriate proportions. In order to confirm the pre-treatment was done properly, it must be verified that the final solution contains the radionuclides of the initial samples. In general, the validation experiment uses 9 elemental standards (Ce, Co, Cs, Fe, Nb, Ni, Re, Sr and U) and standard metal samples, but by melting metal samples completely, validation test is quite straightforward. However, it is important to dissolve it in a transparent liquid state, because sediment may form over time, and that can affect the accuracy of measurement.

The best way to determine the concentrations of all radionuclides contained in the metal wastes is by completely dissolving it. However, the solution obtained through complete dissolution method may contain excessive solutes such as iron and nickel, making it challenging for individual separation during subsequent sample pre-treatment processes. Therefore, a comprehensive approach is crucial rather than a partial one.

Acknowledgements

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Keywords: Permanent disposal, Metal waste, Pre-treatment

Performance Improvement Through Replacement of Chemical Hot Cell Manipulator

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The only chemical hot cell in Korea was facilitated at Radioactive Material Chemical Analysis Facility for analyzing nuclear fuel burn up. The chemical hot cell consists of manipulator, remote tong, pneumatic transfer device, padirac and utility equipment. The manipulator is main equipment that is used for various tasks such as analyzing nuclear fuel burn up, inspecting various equipment, working intermediate level waste and radiation decontamination. The existing manipulator frequently broke down due to deterioration of the main driving parts over time. The booting of manipulator needed replacement to maintain airtightness due to the effects of strong acid and high radiation. In this study, we introduce manipulator replacement procedure and performance improvement through it.

Keywords: Chemical hot cell, Manipulator

Dissolution and Burnup Determination of Irradiated U-Mo Alloy Fuel by a Chemical Method Using Nd Isotope Monitors

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The determination of burnup is a crucial step for assessing nuclear fuel performance. Employing both non-destructive and destructive techniques, researchers aim to quantify the number of fissions per heavy atom (mass ≥ 232) initially present in the fuel.

A destructive method was employed to determine the burnup of U-Mo alloy fuel irradiated in the High-flux Advanced Neutron Application Reactor (HANARO) at KAERI. The dissolution procedure is composed of two-steps for sequential dissolution of cladding material and alloy fuel. The fuel sample can be dissolved completely in 6 M HCl and 14 M HNO₃ - 1 M HF mixture, maintained at 60 ~90°C for 6 hours under reflux conditions. A two-step separation for both the unspiked and spiked sample solutions enabled us to achieve the pure separation of various fission products, including U, Pu, and Nd. The isotopic compositions of U, Pu, and Nd were determined using a thermal ionization mass spectrometer (TIMS) by the isotope dilution mass spectrometric method (IDMS), which employed triple spikes of ²³³U, ²⁴²Pu, and ¹⁵⁰Nd. Through precise analysis and measurement, the burnup of the U-Mo alloy fuel was accurately determined.

Keywords: Burnup, Nd monitor method, U-Mo alloy fuel, IDMS, TIMS

Improvement of Negative Pressure Measurement System in Chemical Hot Cell

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Chemical hot cell must maintain negative pressure to protect the safety of experimenter from internal radioactive dust and harmful gas. A gauge is installed in chemical hot cell to check negative pressure. We can check whether the appropriate negative pressure is maintained through the gauge. Existing differential pressure gauge had the inconvenience of having to be regularly calibrated by outside contractor. To improve this, we replaced the differential pressure gauge with a manometer. In this study, we introduce a manometer replacement procedure and performance improvement through it.

Keywords: Chemical hot cell, Negative pressure, Differential pressure gauge, Manometer

Feasibility Study of Radiochemistry Analysis for Radioactive Metallic Waste

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For the operation of nuclear power plant (NPP), there are several major components that is necessary for nuclear power generation. Steam generator and reactor are one of them. However, just like all other components in NPPs, those components also have designed lifespan to be used in the reactor system for its durability. Thus, after the lifespan they have to be replaced and evaluated for nuclide inventory in accordance with disposal acceptance regulations. The steam generator is made of stainless-steel material to maintain integrity in the high-temperature, high-pressure, and oxidizing environment of the primary system of the reactor, and the reactor head is made of carbon-steel material. To evaluate the applicability of radiochemical analysis methods for these materials, it was decided to apply them to respective standard specimens. The standard specimens were sectioned into small sizes and liquefied using strong acids. The composition table of these specimens was used to check the recovery rate before and after pretreatment. Extraction chromatography was conducted for the iron separation and separation of impurities such as nickel. Based on this study, it is possible to confirm the effect on the separation of nuclide present in small quantities when the concentration of iron is significantly high. Also, with this result, it is expected to be applicable to actual radioactive metal waste.

Keywords: Radioactive waste, Radiochemical analysis, Metallic sample, Radioisotope inventory

A Review of ^3H and ^{14}C Analysis Methods

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To analyze the ^3H and ^{14}C isotopes present in Dry Active Wastes (hereinafter referred to as “DAW”) from nuclear power plants, a wet extraction method can be employed. However, wet extraction methods may face challenges due to the generation of secondary waste, such as sulfuric acid, which is classified as a special management material in Korea. Therefore, we are currently attempting fundamental research to apply thermal decomposition methods in the future. In this study, we have described the pre-reviewed methods for analyzing ^3H and ^{14}C isotopes in preparation for changing the experimental methods.

The analysis of ^{14}C radioisotopes involves simultaneous sample pre-treatment and chemical separation processes. Typically, ^3H isotopes are analyzed concurrently with ^{14}C isotopes. ^3H isotopes are vaporized as tritiated water (HTO), while ^{14}C is oxidized to carbon dioxide ($^{14}\text{CO}_2$) for individual measurement and analysis. The vaporization and oxidation methods primarily include thermal decomposition and wet extraction.

First, thermal decomposition involves heating the sample to absorb HTO and $^{14}\text{CO}_2$ into their respective trapping solutions for liquid scintillation counting (LSC). The analytical equipment for this method includes thermal decomposition devices such as Pyrolyser or Oxidizer. The Pyrolyser is composed of three zones and can be heated up to 800 degrees. The Oxidizer, on the other hand, can process 20~60 samples per hour using less sample volume compared to the Pyrolyser, resulting in shorter analysis times. However, the Oxidizer requires more advanced maintenance efforts than the Pyrolyser. Lim et al. (2015) analyzed ^3H with an efficiency ranging from 77% to 99% and ^{14}C with an efficiency exceeding 94% using thermal decomposition. Additionally, X. Hou (2005) reported efficiencies of over 98% for ^3H and over 95% for ^{14}C in graphite or concrete samples.

Wet extraction involves decomposing the sample with a mixed acid to vaporize ^3H as HTO and oxidize ^{14}C to $^{14}\text{CO}_2$ for measurement via LSC. The apparatus includes sample decomposition containers, reflux devices, and trapping containers. Ahn et al. (2013) achieved approximately 98.2% efficiency in ^3H analysis using wet extraction, while ^{14}C analysis results varied depending on the organic compound ($\text{NaH}^{14}\text{CO}_3$ and $^{14}\text{CH}_3\text{OH}$: over 95%, $\text{C}_6\text{H}_5^{14}\text{CH}_3$: approximately 83.3%).

Analyzing ^{14}C isotopes in biological samples involves homogenizing the samples, freeze-drying to remove moisture, combustion in a sample combustion device, capturing CO_2 with NH_4OH solution, precipitating CaCO_3 with CaCl_2 , acidifying to release CO_2 , and measuring CO_2 in the trapping solution. Atmospheric samples follow a similar procedure after capturing CO_2 with Molecular sieve and combustion at 450 degrees.

This study reviewed existing research on analyzing radioactive ^3H and ^{14}C isotopes. Through these diverse research examples, we can not only apply methods suitable for waste forms but also derive innovative ideas. Future research will focus on experimenting with thermal decomposition methods for DAWs.

Keywords: ^3H , ^{14}C , Thermal decomposition, Wet extraction

Overseas Case Study: Applying the Mean Activity Concentration Method as an Indirect Evaluation for Radioactive Waste Isotopic Inventory

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To dispose of waste generated from nuclear power plants, it is required to evaluate the inventory of radioactive isotopes, which accounts for more than 95% of the radioactivity in the waste, according to the regulations of the Nuclear Safety Commission. Korea Hydro & Nuclear Power (KHNP) uses the scaling factor method (hereinafter referred to as “SF method”) as an indirect method for evaluating the inventory of radioactive isotopes. Additionally, for waste generated before 2003, efforts are being made to apply the mean activity concentration method (hereinafter referred to as “MAC method”) in addition to the SF method.

This study aims to describe cases where the MAC method was applied internationally for the disposal of radioactive waste. Detailed reports applying the MAC method can be found in Japan’s “JAEA-Technology-2015-009” and “JAEA-Technology-2019-015” reports. While it was challenging to find detailed reports or papers from overseas regarding the application of the MAC method, cases from the United States, Japan, Spain, France, and Brazil were confirmed as follows:

- In the United States, a study tracked the ratio of $^{63}\text{Ni}/^{60}\text{Co}$ concentrations in BWR RWCU resin over 20 years, confirming a slope of “0”, similar to the concept of MAC method (Source: IAEA Nuclear Energy Series, No. NW-T-1.18).
- In Japan, the MAC method was applied for evaluating the inventory of H-3 isotopes in solid waste generated during nuclear power plant operation (Source: Hideki Masui, ICEM’ 99).
- In Spain, cases were found where the MAC method was applied for difficult-to-measure isotopes during the decommissioning of the Cabril facility and for evaluating ^{99}Tc isotope inventory (Source: A. Morales, WM’05 Conference).
- In Brazil, for low and intermediate-level waste disposal, the arithmetic means of directly analyzed data from the stream was used when it was challenging to evaluate DTM isotopes using scaling factors (Source: IAEA Nuclear Energy Series, No. NW-T-1.18).

Before applying the MAC method in South Korea, we reviewed overseas cases and confirmed that several countries are already applying this method as an indirect evaluation method based on actual analyzed values.

Keywords: Indirect evaluation, Mean activity concentration, Scaling factor, Isotopic inventory

Construction of Radionuclide Analysis Laboratory as Test-Bed for Decommissioning Wastes at KRID

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Korea Research Institute of Decommissioning (KRID) was established in 2020 to decommission commercial nuclear power plants (NPPs). we will study and support to entire nuclear decommissioning process, which includes decontamination and cutting of structures within NPPs such as reactor vessels (RVs) and reactor vessel internals (RVIs), as well as waste management, radionuclide analysis, and site restoration. KRID are being built in Busan and Gyeongju, respectively, near the KORI#1 and WOLSONG#1 NPPs, which are scheduled for decommissioning. Among these, buildings for non-radiation work such as buildings of office, research, and mock-up are scheduled to be completed in Busan untill 2024. And the radiation work building for characterization analysis of decommissioning wastes will be completed in 2026-2027.

Additionally, we have designed a test-bed laboratory for radionuclide analysis of decommissioning wastes in a research building to the support the timely decommissioning of the NPPs. It will be used to develop, verify and improve of KRID's radionuclide analysis procedures each nuclides and matrix, and radionuclide analysis will be performed. In this paper, we are report the design, contruction plan, analysis scenario in radionuclide analysis laboratory as test-bed. And we will also plan to accept license about radiation experiment for radionuclide analysis of decommissioning wastes in the research building.

Keywords: Decommissioning wastes, Radionuclide analysis, Radiation safety, R&D project, KRID

Establishment of a Quality Management System Plan for Rapid and Reliable Radioactivity Analysis of Decommissioning Waste at KRID

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The need to establish a systematic quality management system for stable and efficient management of decommissioning waste and rapid response to the demand for radioactivity analysis is becoming important. Since the establishment and validation of radioactivity analysis infrastructure is necessary to meet the demand for waste disposal generated during the pre-decommissioning process.

Accordingly, KRID (Korea Research Institute of Decommissioning) is seeking to establish a systematic quality management system for accurate radioactivity analysis of decommissioning waste. The main factors of a quality management system are experienced analysts, continuous training, optimized facilities and organized documentation. As a result of radioactivity analysis that has been proven to be effective based on KOLAS-G-009 standards, we intend to promote the decommissioning of nuclear power plants with the priority of accuracy, safety and compliance.

Keywords: Radioactivity analysis, Validation, Quality management system, Decommissioning wastes

Development of Pretreatment Technique for Analysis of Uranium in Simulated LiCl-KCl- UCl_3 Molten Salt Samples

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High-precision analysis techniques for spent nuclear fuel are necessary to strength nuclear non-proliferation and ensure the feasibility of implementing pyro safeguards. Since spent nuclear fuel contains various elements, pretreatment processes such as element screening are important prior to analysis for efficient analysis. In this study, a method for screening elements in molten salt samples was developed by means of TXRF (Total-reflection X-ray Fluorescence), non-destructive analysis technique. Using LiCl-KCl molten salt, three types of simulated molten salt samples were prepared by adding 1.3wt%, 0.13wt%, and 0.013wt% of UCl_3 to the molten salt. For various precise chemical analyzes of prepared molten salt samples, it is necessary to first dissolve the solid sample. It is preferable that the amount of sample collected is as small as possible in terms of waste reduction. For this purpose, a small of molten salt sample of 0.1 g was collected and dissolved into acid solution. The solutionized sample was loaded into the center of the carrier for TXRF measurement, dried completely, and then mounted on the TXRF device and measured. As a result, U, Ce, and Nd peaks along with K were confirmed. These peaks were due to the elements that were added at more than 1wt% when preparing the molten salt samples, and it was confirmed that the molten salt samples could be screened through TXRF analysis.

Keywords: LiCl-KCl- UCl_3 molten salt, Uranium, Pretreatment, Elemental screening, TXRF

The Separation Method of Steel Samples by Ketone Separation

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Currently, the non-volatile nuclides ^{94}Nb , ^{99}Tc , ^{90}Sr , ^{55}Fe and $^{59/63}\text{Ni}$ are separated by sequential separation. The non-volatile nuclides ^{94}Nb , ^{99}Tc , ^{90}Sr , ^{55}Fe and $^{59/63}\text{Ni}$ were separated by ketone separation to remove Fe^{2+} from the sample prior to conventional separation. The ketone separation method used diisobutyl ketone solution separation and after dissolving the iron samples, we found that over 90% of the Fe^{2+} was separated by ketone separation. The ketone separated solution was applied to a conventional separation method and the recoveries were checked by ICP-MS and ICP-OES and found to be over 70%, confirming that non-volatile nuclides can be separated by ketone separation.

Keywords: Separation, Steel, Ketone

Laser-Induced Breakdown Spectroscopy (LIBS) Measurements of Strontium Concentrations in FLiNaK Molten Salts

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Molten Salt Reactors (MSRs) are becoming increasingly recognized as a prominent feature of fourth-generation nuclear reactor technology due to their high energy efficiency and enhanced safety. However, one distinct characteristic of MSRs is the homogeneous mixing of the fuel and coolant, which results in the accumulation of fission products, rare earth elements, and actinium fluorides in the molten salt fuel and the subsequent alteration of the fuel properties over time. For that reason, the real-time monitoring of the fuel composition is critical for the reactor's operational efficiency and safety management.

In this study, we investigated the application of Laser-Induced Breakdown Spectroscopy (LIBS) as a practical tool for the real-time detection of strontium (Sr), a prominent fission product with high fission yield, in FLiNaK molten salts. The experiments were performed in a laboratory-scale furnace under an argon atmosphere, whereby 532 nm Nd:YAG laser pulses were focused on the surface of the molten salt sample placed in glassy carbon (GC) crucibles, and the corresponding emitted signal of plasma light were captured by an Echelle spectrometer. The accuracy and precision of the in-situ LIBS method was verified by comparing it against data obtained from the conventional ex-situ analytical technique of Inductively Coupled Plasma Mass Spectrometry (ICP-MS) and solid-state LIBS measurements, both obtained through molten salt sampling.

Our results yielded calibration curves spanning a broad Sr concentration range of thousands of ppm and led us to conclude that the LIBS method can achieve a lower Limit of Detection (LOD) than the actual Sr concentrations found in molten salt reactors. These results highlight the quantitative capabilities of the LIBS sensor to significantly enhance MSR safety and operational efficiency through the real-time monitoring of fission product concentrations in MSR fuels.

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Keywords: Molten salt analysis, Fission products, Laser-induced breakdown spectroscopy, Strontium, FLiNaK

Colorimetric Detection of Phenol and H₂O₂ by a New Peroxidase-Like Copper-based Metal-Organic Framework

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In this study, we present a newly synthesized peroxidase-like copper-based metal-organic framework (MOF) and investigate its potential for colorimetric sensing of phenol and hydrogen peroxide (H₂O₂). The MOF, characterized by its unique structure and composition, demonstrates remarkable sensitivity and selectivity towards both analytes. Utilizing simple colorimetric assays, we demonstrate the capability of the MOF to detect phenol and H₂O₂ with high precision and efficiency. Furthermore, the MOF displays excellent stability across a wide pH range of 3 to 12, coupled with remarkable reusability, rendering it a promising candidate for diverse applications in environmental monitoring, biomedical diagnostics, and chemical sensing. Overall, our findings underscore the utility of this peroxidase-like copper-based MOF as a versatile platform for sensitive and reliable colorimetric sensing.

Overall, our findings highlight the potential of this novel copper-based MOF as a versatile platform for colorimetric sensing in various fields. The photocatalyst was characterized using single X-ray diffraction (SC-XRD), powder X-ray diffraction (PXRD), Fourier Transform Infrared Spectroscopy (FTIR), and UV-vis diffuse reflectance spectroscopy (DRS), UV-vis spectrophotometer (UV).

Keywords: Metal-organic framework, Phenol, H₂O₂ sensing, Peroxidase-like MOF

Feasibility Study of Sample Amounts and Pretreatment Method for Spent Resin Sample Nuclides Analysis

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The radioactive waste generated from nuclear power plants must be improved in accordance with the disposal requirements required by regulatory agencies and disposal facilities. In order to ensure disposal safety, it must be reduced, treated, and packaged in a form suitable for disposal. Among the various types of radioactive waste generated, such as concentrated waste liquid, spent resin, sludge, are classified as homogeneous waste. Homogeneous waste must be solidified, followed by tests for compressive strength, irradiation, leaching, submersion, and thermal cycling tests. However, this is not the case when high-integrity containers.

Korea Hydro & Nuclear Power has decided to disposal homogeneous waste using high-integrity containers. Currently, approval for the 860L polymer concrete high-integrity container (PC-HIC) for the disposal of low-dose spent resin and concentrated waste liquid has been completed. Additionally, a disposal plan for the spent resin, which is the subject of initial disposal using the 860 L PC-HIC, has been established in line with the completion date of the second phase construction project of the KORAD. Consequently, We attempted to confirm the pretreatment method for spent resin samples to identify nuclides, which is one of the evaluation items for radioactive waste characteristics. In order to minimize the exposure of the analyst, a comparative experiment of pretreatment method was performed, and the optimal amounts of sample for spent resin analysis was determined.

Keywords: Radioactive waste, Radiochemical analysis, Spent resin, Disposal

Spatial Distribution of Metallic Precipitates in a Spent Fuel Assembly

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Metallic precipitates, contained in spent fuel assembly, are fission products characterized by their materials as noble metals. These phases typically occur as minute particles ranging from nanometers to micrometers in size, situated within the grain boundaries of the fuel material. Given their placement amidst the grain structure, they can be regarded as structural imperfection within the fuel pellet. Notably, the thermal expansion characteristics of fuels containing metallic precipitates tend to be marginally higher compared to those without. This becomes particularly pertinent during elevated temperature conditions post-reactor discharge, where inadequate cooling measures can potentially lead to fuel failure.

In assessing the impact of metallic precipitates on post-reactor fuel integrity, attention must be given to their concentration levels. Utilizing the SCALE/ORIGEN code, an evaluation of metallic precipitates behavior at the fuel assembly scale was conducted, alongside an analysis of their radial distribution within the assembly. The radial power distribution of nuclear reactor was determined utilizing SCALE/TRITON, with consideration given to the Westinghouse 17×17 configuration. The axial power distribution was also calculated using SCALE/TRITON and some references of reactor physics. The axial distribution was divided into 18 zones along the height of the fuel assembly. By integrating the radial and axial power distributions, a three-dimensional SCALE/ORIGEN analysis was performed and the spatial distribution of metallic precipitates concentrations was finally obtained.

Keywords: Used fuel, Metallic precipitates, High-level waste, Grain boundaries

The Separation Method of ^{55}Fe and ^{63}Ni by a Carbonate Precipitation

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Carbonate precipitation was used to separate Fe-55 and Ni-63 from the non-volatile nuclides Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, and Tc-99 using a different sequential separation method. After dissolving the initial sample, Na_2CO_3 was added to identify the precipitate. The precipitate showed that more than 90% of the initial sample was Fe and Ni. After separating Fe-55 and Ni-63 by conventional methods, we checked the recovery by ICP-MS and found that more than 90% of Fe and 80% of Ni were recovered from the precipitate.

Keywords: Separation, Steel, Nickel, Carbonate precipitate

Solubility Study of $\text{NpO}_2(\text{OH})(\text{am})$ in Oxidative KURT Groundwater Condition

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It is necessary to understand the physicochemical behavior of neptunium in the domestic groundwater condition for the long-term safety assessment of spent nuclear fuel disposal. The relevant oxidation states of Np under groundwater conditions are +4 and +5. In this study, Np(V) stock solution and solid phase samples were prepared and the solubility and complexation reactions of Np(V) were studied in domestic groundwater conditions. The dissolution of $^{237}\text{NpO}_2(\text{s})$ in mixed acids produced a mixture of hexavalent NpO_2^{2+} and pentavalent NpO_2^+ . Reduction to Np(V) was conducted by addition of drops of H_2O_2 into the stock solution. The final Np(V) stock solution was green in color and showed a strong absorption at 980 nm. Completion of the reduction reaction and the concentration of the stock solution were determined by absorption spectrum measurements. Neptunyl hydroxide, $\text{NpO}_2(\text{OH})(\text{am})$, solid sample was prepared by adding NaOH solution to the Np(V) solution in an Ar-filled glovebox. Solubility of the solid phase was measured in synthesized oxidative KURT-DB3 groundwater. Dissolved Np concentrations were monitored periodically over 158 days by LSC measurements after ultrafiltration (10 kDa). In equilibrium, Np concentration was measured to be 1.67 ± 0.07 mM at pH 8.27. In the end of the solubility experiment, residual precipitations and supernatants were analyzed by XRD and absorption spectroscopy, respectively. The measured solubility and aqueous chemical species distributions were comparable to those predicted by geochemical modeling based on ThermoChimie (11a) database.

Keywords: Pentavalent neptunium, Neptunyl hydroxide, Solubility, KURT groundwater, Absorption spectroscopy, Amorphous solid

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