

2022 **추계** 학술발표회 논문요약집

2022. 10. 26(수)~28(금) 제주 신화월드 랜딩컨벤션센터



KOREAN
RADIOACTIVE WASTE
SOCIETY
AUTUMN 2022



사단 **한국방사성폐기물학회**
법인 **Korean Radioactive Waste Society**

주최/주관

한국방사성폐기물학회

후 원 사

(주)세아베스틸

한국과학기술단체총연합회

한국원자력협력재단

제주컨벤션뷰로

(주)비츠로넥스텍

(주)선광티앤에스

(주)에스이

한국원자로감시기술(주)

페이브텍 주식회사

(재)사용후핵연료관리핵심기술개발사업단

(주)코네스코퍼레이션

지에스중공업(주)

(주)세종이엔씨

(주)엘림글로벌

(주)태린

(주)큐라캠

주식회사 라드솔

(주)아이티캠

2022 추계 학술발표회논문요약집

This work was supported by the Korean Federation of Science and Technology Societies(KOFST) grant funded by the Korean government.

2022 추계 학술발표회 논문요약집

2022. 10. 26(수)~28(금) 제주 신화월드 랜딩컨벤션센터



**KOREAN
RADIOACTIVE WASTE
SOCIETY
AUTUMN 2022**



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

Contents

KOREAN
RADIOACTIVE WASTE
SOCIETY

1분과 핵주기정책·규제 및 비확산 [ORAL]

- 1-0-01 A Study on North Korea's Denuclearization Policy Suggestion Through Ukraine–Russia War Analysis 37
Sungmin Yang, Gyunyoung Heo [Kyung Hee Univ.]
- 1-0-02 Development of Energy Mix Simulation Model for 2050 Net–Zero Carbon .. 38
Pilhyeon Ju, Sungyeol Choi, Jongho Lee [Seoul Natl. Univ.]
- 1-0-03 Development of Sample Size Calculation Code for Review of IAEA Inspection 39
Byung Hee Won, Hyun–Jo Kim, Dae–Yong Song, Bong Young Kim, Ju–Ang Jung [KAERI]
- 1-0-04 Development and Validation of Prototype Real–Time Verification Sealing System 40
Jihyun Ahn, Hee Seo [JeonBuk Natl. Univ.]
- 1-0-05 Study on the Applicability of Event Tree and Fault Tree Analyses to the Systematic Generation of Diversion Paths 41
Seong–Kyu Ahn, Bong Young Kim, Dea–Yong Song, Ho–Dong Kim [KAERI]
- 1-0-06 An Analysis on Security Activities of the NuScale Standard Design Review .. 42
Kookheui Kwon, Seongyoun Jo [KINAC]
- 1-0-07 A Review on the DBT and Nuclear Damage Compensation Act 43
Sangcheol Hyung, Jaeyeong Jang [KINAC]

1분과 핵주기정책·규제 및 비확산 [POSTER]

- 1-P-01 A Study on the Analysis of Design Features of SMART to Apply Safeguards–By–Design 47
Joo Hyung Moon [KAERI], Donghyuk Lim [KINAC]
- 1-P-02 Investigation of Physical Protection System Design on Unite States Standards and Practices 48
Jihawn Cha [KINAC]

1-P-03	Application of the Facility Safeguardability Assessment Process to the Decommissioning of Nuclear Facilities in the ROK	49
	Seungmin Lee, Donghyuk Lim [KINAC]	
1-P-04	Analysis of Cyber Security Regulatory Frameworks for EP CDAs in the U.S. ..	50
	In-hyo Lee [KINAC]	
1-P-05	A Study on the Changes of Safeguards Techniques and Equipment	51
	Inchul Kim, Hyunjo Kim, Byungdoo Lee, Sungho Lee, Jooang Jung, Jinha Choi, Munyung Ryu, Seongmi Han [KAERI]	
1-P-06	A Study on the Establishment of Integrated Database for Safeguards and Nuclear Export Control	52
	Sungho Yoon [KINAC]	
1-P-07	A Study on Identification of Vital Areas in Spent Fuel Storage Facilities to Protect Against Sabotage	53
	Minho Kang, Sundo Choi [KINAC]	
1-P-08	A Localization Strategic Study on the Export Control Programs of Sweden, Germany, France and the United Kingdom	54
	Jae Kwang Kim, Seunghyo Yang [KINAC]	
1-P-09	A Study on the Export Control Implementation in Korea Considering the Updated EU Regulation 821/2021 for Dual-use Items Transfer Control	55
	Jae Kwang Kim, Seunghyo Yang [KINAC]	
1-P-10	A Study on Supply Chain Control Regulation of Nuclear Facilities by SSEP Functions	56
	Seunghoon Park, Poe il Park, Kookheui Kwon, Chae-Chang Lee [KINAC]	
1-P-11	A Study on the Structure Response of Explosion Simulation Models for the Evaluation of Explosive Threats to Nuclear Power Plants	57
	Eojin Jeon, Wooseub Kim, Hyeseung Kim, Sundo Choi [KINAC]	
1-P-12	A Study on I&C System Decision Process for Cyber Security Test	58
	In-hyo Lee [KINAC]	
1-P-13	A Study on the Improvement of X-ray Security Search Education for Physical Protection Personnel	59
	Donghyun Kim [KINAC]	
1-P-14	A Study on the Development of the Curriculum for “Disguised Weapon Identification” and “Abnormal Behavior Detection Method”	60
	Donghyun Kim [KINAC]	

1-P-15	A Methodology for Deriving Safety Boundaries of Nuclear Power Plants Against Unmanned Aerial Vehicle Attacks	61
	Seokwoo Sohn, Hyeseung Kim, Sundo Choi [KINAC]	
1-P-16	A Study on the Export Control Policy of the U.S. on the Conditions of IAEA Additional Protocol for Nuclear Power Plant Export	62
	Seung-hyo Yang [KINAC]	
1-P-17	Analysis and Prospect of the Implementation of Random Inspection in ROK	63
	Chul Heo, Donghyuk Lim [KINAC]	
1-P-18	A Study on the Measurement Uncertainty for Evaluation of Material Balance in Bulk Handling Facility	64
	Yewon Kim, Haneol Lee [KINAC]	
1-P-19	Development of Evaluation Software for Security System Effectiveness	65
	Hyeseung Kim, Sundo Choi, Sung Soon Jang [KINAC]	
1-P-20	Study on Measures to Improve the Quality of Nuclear Regulation by Regulatory Authority	66
	Jaeyeong Jang [KINAC]	
1-P-21	Analysis of North Korea's Foreign Trade and Logistics Infrastructure	67
	Han-sol Ko, Chan-suh Lee, Seung-hyo Yang [KINAC]	
1-P-22	A Review on the Interpretation of EDP Term for Trigger List Items	68
	Chansuh Lee, Hansol Ko, Seunghyo Yang [KINAC]	
1-P-23	Development of Semi-Automatic Satellite Imagery Analysis System to Support Monitoring of Potential Nuclear Activities	69
	Gayeon Ha, Jae-Jun Han, Minsoo Kim [KINAC]	
1-P-24	A Review on the Management System for Items Subject to a Nuclear Cooperation Agreement Owned by Nuclear Licensees	70
	Hee Su Choe, Seung-hyo Yang [KINAC]	
1-P-25	The Analysis of Major Priliferation Resistance Barriers in the Dry-processing of Spent Fuels for an Efficient Final Disposal	71
	Won Il Ko, Hyo On Nam, Yung-Jun Cho [KAERI]	
1-P-26	A Study on the Establishment and Utilization of National Nuclear Technology Information Collection and Analysis System	72
	Hyun-doo Kim, Seung-hyo Yang [KINAC]	
1-P-27	The Requirements Analysis and Concept Design for Development of Obligated Items Management System	73
	Mun-Young Ryu, In-Chul Kim, Seong-mi Han [KAERI]	

1-P-28	Development of Installation Standards and Evaluation Indicators for Passive Infrared Sensor in National Security Facilities	74
	Hyeseung Kim, Sundo Choi [KINAC]	
1-P-29	A Study on the Response for Drone Terrorism in Nuclear Facilities by Analyzing Statistical Data of Drone Incidents	75
	Sundo Choi, Hyeseung Kim, Seokwoo Sohn [KINAC]	
1-P-30	A Study on the Performance Test of Biometrics for Counterfeit Technology in Nuclear Facilities	76
	Sundo Choi, Hyeseung Kim [KINAC]	
1-P-31	Using Textual Analysis of News Articles to Identify State's Nuclear Activity	77
	Hojung Do, Dongjin Kim, Minsoo Kim [KINAC]	
1-P-32	A Study on the Effect of Frequency Scaling Factor on FBP	78
	Jisu Kim, Jihun Moon, Heejun Chung, Sung Woo Kwak [KINAC], Kyung Taek Lim [Sejong Univ.]	
1-P-33	A Study on the Analysis of Research Outputs of KINAC (2010–2021) and Support Methodology for Performance Diffusion	79
	Hyun Young Kim, Heejun Chung [KINAC]	
1-P-34	The Necessity of Establishing a Management System for Obligated Items Subject to Bilateral Cooperation Agreement	80
	Seongmi Han, In-Chul Kim, Mun-Young Ryu [KAERI]	
1-P-35	Monte Carlo Simulation for Designing a CZT-based Fork Detection System ...	81
	Wooseong Hong, Geehyun Kim [Seoul Natl. Univ.]	
1-P-36	Analysis of Nuclear Material Distribution Behaviors Under the Abnormal Conditions of the Nuclide Management Technology	82
	Bong Young Kim, Young Eun Jung, Se-Hwan Park, Seong-Kyu Ahn [KAERI]	
1-P-37	A Study on the Application of Safeguards to Small Modular Reactor	83
	Jinha Choi, Hyun-Jo Kim, Byung-Doo Lee, Sung-Ho Lee, Juang Jung [KAERI]	
1-P-38	Development of Process Flow Simulator of Uranium Fuel Fabrication Facility for Material Balance Evaluation Analysis	84
	Byung Hee Won, Hee-Sung Shin, Ho-Dong Kim, Jinha Choi [KAERI]	
1-P-39	Evaluation of Blast Loads on a Reinforced Concrete Wall Under Internal Explosion	85
	Wooseub Kim, Hyeseung Kim [KINAC]	
1-P-40	A Study on Request of Governmental Assurance for the Export of Trigger List Items	86
	Su-Hyeon Kim, Seung-hyo Yang [KINAC]	

1-P-41	A Study on the License for Intermediate Trade and Merchandising Trade of Trigger List Items	87
	Su-Hyeon Kim, Han-Sol Ko [KINAC]	
1-P-42	Considerations for Nuclear Power Plant Cyber Security Contingency Plan ...	88
	Seungmin Kim [KINAC]	
1-P-43	Estimation of U Ore Production Using Bayesian Network	89
	Yonhong Jeong, Hojung Do [KINAC]	

2분과 사용후핵연료 처분전관리 [ORAL]

2-0-01	Machine Learning Based Quality Control for Spent Fuel Safety Information ..	93
	Ga-hee Sim, Moon-Ghu Park, Kyoon Ho Cha [Sejong Univ.]	
2-0-02	Electrochemical Impedance Spectroscopy Study of Hastelloy N in Molten LiCl-KCl Salt at 500°C	94
	Jun Woo Park, Jong-II Yun [KAIST]	
2-0-03	Separation of Sr From Simulated Oxide Fuel Using Molten Chloride Salts	95
	Eun-Young Choi, Sang-Kwon Lee, Seungwoo Park, Hamin Choi, DongHyun Jeon, Chang Hwa Lee [KAERI], Minsu Lee [UST]	
2-0-04	Investigation of Radial Hydride Effect on Mechanical Properties of Zircaloy Cladding Via Hydride Morphology Image Analysis	96
	Dahyeon Woo, Youho Lee [Seoul Natl. Univ.]	
2-0-05	A Study on the Properties of B ₄ C/Al Composites for Absorbing Thermal Neutrons Manufactured by the Stir Casting Process	97
	Donghyun Lee, Seungchan Cho, Junghwan Kim, Sang-bok Lee, Sang-kwan Lee [KIMS]	
2-0-06	Evaluation of Physicochemical Characteristics of Group II Nuclide Ceramic Waste Form for Long-term Storage and Disposal	98
	Byeonggwon Lee [Kyungpook Natl. Univ.], Jung-Hoon Choi, Ki Rak Lee, Hyun Woo Kang, Hwan-Seo Park [KAERI]	
2-0-07	An Analysis of Neutron and Gamma Sources in Spent Fuels Using SCALE-ORIGEN ARP	99
	Sohee Cha, Kwangheon Park [Kyung Hee Univ.]	
2-0-08	Development of Model to Evaluate Thermal Fluid Flow Around a Submerged Transport Cask of Spent Nuclear Fuel in the Deep Sea	100
	Guhyun Jeoung, Sungyeoun Kim, Sanghoon Lee [Keimyong Univ.]	

2분과 사용후핵연료 처분전관리 [POSTER]

2-P-01	Spent Fuel Characteristics of Small Modular Reactors and Its Management Implications	103
	Dahyeon Woo, Chansoo Lee, Youho Lee [Seoul Natl. Univ.]	
2-P-02	Analysis of Hydrogen Migration in Zircaloy Cladding During Dry Storage by Simulating Hydrogen Diffusion and Phase Change Kinetics	104
	Chansoo Lee, Youho Lee [Seoul Natl. Univ.]	
2-P-03	Development of a Radial Hydride Fraction Prediction Model and Experimental Validation	105
	Changhyun Jo, Youho Lee [Seoul Natl. Univ.]	
2-P-04	Comparison of Fire Test Using Slice Model and Half-scale Model of the KTC-360 Transport Cask	106
	Kyoung-sik Bang, Yun-Young Yang, Woo-Seok Choi [KAERI]	
2-P-05	Suggestion of Simple Rod Internal Pressure and Free Void Volume Correlations in Spent Fuel Rod	107
	Yongsik Yang, Donghak Kook [KAERI]	
2-P-06	Analysis of the Regulatory Processes for Spent Nuclear Fuel Transportation Cask	108
	Hyun Jun Na, Hyeok Jae Kim, Min Seong Kim, Kwang Pyo Kim [Kyung Hee Univ.]	
2-P-07	Decay Heat Estimation of Spent Fuel Assemblies for Decommissioning of Kori Units 3 and 4	109
	Wonjune Mah, Chang-Lak Kim, Sun-kee Lee [KINGS]	
2-P-08	Examination for Evaluating Degradation of Guide Tubes in Spent Fuel Assembly	110
	Myungchul Park, Sunggeun Kim, Jaehan Kim, Sukwoo Hong, Hyojik Lee [KAERI]	
2-P-09	Introduction of Spent Fuel Information Management Model Based on Semantic Web Technology	111
	Hong-jin Kim, Young-Chul Cho, Kyoong-Ho Cha [Sejong Univ.]	
2-P-10	Effect of Li_2O Addition on Precipitation Behavior of Uranium in LiCl-KCl-UCl_3 Salt	112
	Taeho Kim, Su Jeong Heo, Dalsung Yoon, Seungwoo Paek, Chang Hwa Lee [KAERI]	
2-P-11	Separation of Salt From Strontium Carbonate and Uranium Oxide Powders	113
	Sang Woon Kwon, Su Jeong Heo, Joon Bo Shim, Chang Hwa Lee [KAERI]	

2-P-12	UO ₂ and SIMFUEL Pellets Fabrication and Leaching Behavior Study for Stabilization Technology of Damaged Used Fuel	114
	Su Jeong Heo, Ju Ho Lee, Jae-Won Lee, Yung-Zun Cho [KAERI]	
2-P-13	A Study on Optimization of UCl ₃ Production by Reaction of Uranium Metal and Ammonium Chloride	115
	Dalsung Yoon, Seungwoo Paek, Sang-Kwon Lee, Ju Ho Lee, Taeho Kim, Chang Hwa Lee [KAERI]	
2-P-14	Fluidization Bed Behaviors of Uranium Oxide Powder Using Surrogates	116
	Jae Won Lee, Ju Ho Lee, Sun Seok Hong, Yung zun Cho [KAERI]	
2-P-15	Solvent Effect on Properties of Ag-Containing Aluminosilicate for Capturing Iodine	117
	Jae Won Lee, Ju Ho Lee, Chang Hwa Lee, Yung Zun Cho [KAERI]	
2-P-16	Improvement of the Evaluation Method for the Radioactivity in Spent Nuclear Fuel	118
	Jongho Hong, Sangil Ahn, Jaechoon Lee [KHNP], Kiyoun Kim [KHNP-CRI]	
2-P-17	Discretization Uncertainty Estimation for Thermal Analysis of the Spent Fuel Transport Cask	119
	Sung-yong Lee, Dong-gyu Lee [KONES Corp.]	
2-P-18	~~Spent Fuel Pool Criticality Evaluation by Applying Boron Credits~~	~~120~~
	~~Yun-Sik Kim, Dong-Gyu Lee [KONES Corp.], Ki-Young Kim [KHNP-CRI]~~	
2-P-19	The Quantification of Residual Water by Using the Drying Process Data in Dry Storage System	121
	Geon-Hui Lee, Nam-Hui Lee, Kyung-Wook Shin, Gyung-Sun Chae [SAE-AN Engineering Corp.], Seung-Hwan Yu [KAERI]	
2-P-20	Criticality Impact Assessment of Fuel Rod Loss for the Development of Spent Fuel Acceptance Criteria	122
	Janghwan Lim, Seongki Lee, Jongsung Yoo [KEPCO NF]	
2-P-21	Assessment of Potential Degradation Mechanisms to Resolve Integrity Issues of CANDU Spent Fuel	123
	Kang Moon Lee, Nam Ho Lee, Seong Ki Lee, Jong Sung Yoo [KEPCO NF]	
2-P-22	Ring Tensile Test Material Properties Calibration Using Optimization Technique	124
	JaeYong Kim, HongRyoul Oh [KAERI], SungHoon Park [V-ENG]	
2-P-23	Review on Non-Destructive Burnup Evaluation Method for Spent Nuclear Fuel ..	125
	Daegwon Eun, Jae-Hun Ko, Moonoh Kim, Dong-Gyu Lee [KONES Corp.]	

Withdrawal

2-P-24	Applicability Evaluation of Ramberg–Osgood Equation as New Cladding Plastic Material Model	126
	JaeYong Kim, YongSik Yang [KAERI], JuYeop Park [KINS]	
2-P-25	Estimation Model of Residual Water Evaporation During Vacuum Drying	127
	Seunghwan Yu, Kyoung-sik Bang [KAERI]	
2-P-26	Development of Damaged Fuel Canister for Dry Storage	128
	Seo-Jeong Lee, Kwang-Ho Lee, Eun-Ki Lee [KHNP-CRI]	
2-P-27	Quantitative Analysis of Sr and Cs in LiCl–KCl Molten Salt by Laser-Induced Breakdown Spectroscopy (LIBS)	129
	Hyeongbin Kim, Wonseok Yang, Sungyeol Choi [Seoul Natl. Univ.], Yunu Lee [KAIST]	
2-P-28	Seismic Analysis of Concrete Pad in the Dry Storage Facilities According to Soil Properties	130
	Seungpil Kim, Sang Soon Cho [KAERI]	
2-P-29	DeCART2D/MASTER Core Follow Calculation for Hanbit Unit 3 and Comparison With McCARD Single Fuel Assembly Burnup Analyses	131
	Jeong Woo Park, Seung-Ah Yang, Ho Jin Park [Kyung Hee Univ.]	
2-P-30	Thermal Performance Sensitivity Analysis for Modeling of Spent Fuel Assembly in Transport System	132
	Kun-ho Chun, Jung-yeul Lee, Seong-ki Lee [KEPCO NF]	
2-P-31	A Study on the Design Parameters of the Off-gas Trapping System	133
	Byungsuk Park, Seokmin Hong, Juho Lee, Jaewon Lee, Changhwa Lee [KAERI]	
2-P-32	Lessons Learned in Improvement Work of Fuel Handling Equipment at Operating Nuclear Power Plants	134
	Sang-Gyoon Chang, Won-Bae Park, Min-Gyu Kim [KEPCO E&C], Dae-Hyuk Kwon [KHNP]	
2-P-33	A Preliminary Consideration to Categorize Classes of Major SSCs of a Dry Storage System for SNF Based on Importance to Safety	135
	Sang Hyeok Lee, Jae Hak Cheong [Kyung Hee Univ.]	
2-P-34	Effect of Salt Stirring on Phosphorylation of Uranium and Rare Earths in Molten Salt	136
	Joon-Bo Shim, Hansoo Lee, Chang Hwa Lee [KAERI]	
2-P-35	Experiment for a Collection Method of Phosphorylation Products From Molten Salt	137
	Joon-Bo Shim, Ju Ho Lee, Sang-Woon Kwon, Chang Hwa Lee [KAERI]	

2-P-36	Development of Code for Pressure Resistance Evaluation of Spent Nuclear Fuel Transport Cask Submerged in the Sea	138
	Jiyoung Park, Hakyong Jang, Sanghoon Lee [Keimyong Univ.]	
2-P-37	A GoldSim Model for Evaluating the Effectiveness of Radionuclide Management Process on Environmental Burden Reduction	139
	Hong Jang, Hun Suk Im, Hyo On Nam, Won Il Ko, Yong Zun Cho [KAERI]	
2-P-38	Strategic Study on Various Optional Treatment Processes for Disposal of Spent Nuclear Fuel	140
	Hun-Suk Im, Hong Jang, Hyo-On Nam, Won-Il Ko, Yong Zun Cho [KAERI]	
2-P-39	Preliminary Assessment of Radiation Effects of Spent Fuel Transport to Interim Storage	141
	Yunhui Lee, Chang-Lak Kim, Sanghwa Shin [KINGS]	
2-P-40	Corrosion Behavior of Neutron Absorber Borated Stainless Steel in Simulated Spent Fuel Pool Environment	142
	Daehyeon Park, Yunju Lee, Junhyuk Ham, Seung Chang Yoo, Ji Hyun Kim [UNIST], Kiyong Kim, Donghee Lee, Yongdeog Kim [KHNP-CRI]	
2-P-41	Efficient Radiation Shielding Evaluation for Spent Fuel Storage Facilities Using a Simple Model	143
	Jae Hun Ko, Young Hwan Choi, Dong Gyu Lee [KONES Corp.]	

3분과 고준위폐기물처분 [ORAL]

3-O-01	Corrosion Behavior of Potential Materials for Canister Fabricated Using 3D Printing Methods	147
	Cheong Won Lee, Young-Ho Lee, Sung Chan Yoo [KAERI]	
3-O-02	Geochemical Simulation of U(VI) Sorption on Na-Montmorillonite: Effects of pH, Temperature, Competing Cations, U(VI) Concentration, and Carbonate Species	148
	Jeonghwan Hwang, Jung-Woo Kim [KAERI], Weon Shik Han, Won Woo Yoon [Yonsei Univ.]	
3-O-03	Evaluation of Deep Geological Characteristics Using Deep Boreholes	149
	Dae-Sung Cheon, You Hong Kihm, Kwangmin Jin, Wonkyong Song, Seong Kon Lee, Sung Phil Hyun, Heejun Suk [KIGAM]	
3-O-04	Preliminary Experimental Results for Diffusion of Sr, Sm, and Eu in a Compacted Ca-Bentonite	150
	Nak Kyu Kim, Chung-Kyun Park, Jae-Kwang Lee, Seonggyu Choi, Jang-Soon Kwon [KAERI]	

3-0-05	Preliminary Study for Estimating the Effect of Earthquake-induced Stress on Hydraulic Properties of Fractures	151
	Yong-Min Kim, Jung-Woo Kim [KAERI]	
3-0-06	Inversion of Barcelona Basic Model Parameters of Kunigel-V1	152
	Yeongju Kim, Hoonyoung Jeong [Seoul Natl. Univ.], Changsoo Lee, Minseop Kim [KAERI]	
3-0-07	Analyses on Thermal Stability of the Disposal Systems for High Burn-up PWR Spent Nuclear Fuel in Korea	153
	Gisu Lee, Myeonggi Cha, Namhyeong Kim, HangJin Jo [POSTECH]	
3-0-08	Methodologies to Establish Site-Specific FEP for High-Level Radioactive Waste Repository in Korea	154
	Donghoon Baek, Kibeom Son, Jaehyeon Yang, Haeram Jung, Karyoung Choi, Gyunyoung Heo, Hyungdae Kim, Kunok Chang [Kyung Hee Univ.]	

Withdrawal

3분과 고준위폐기물처분 [POSTER]

3-P-01	Characteristics of AE Wave Propagation Through Concrete Silo of Gyeongju Radioactive Waste Repository Under the Effects of Groundwater and Temperature	157
	Jin-Seop Kim, Chang-Ho Hong, Ji-Won Kim [KAERI], Hang-Lo Lee [Korea Expressway Corporation]	
3-P-02	Factors Controlling Uranium Level in Granitic Aquifer	158
	YeoJin Ju, Ji-hun Ryu [KAERI], Vikram Thapar [Cornell Univ.]	
3-P-03	Spectroscopic Study of Interaction Between U(VI) and Natural Organic Matters in Domestic Groundwater	159
	Euo Chang Jung, Min-Hoon Baik, Dawoon Jeong, Ji-Hun Ryu [KAERI]	
3-P-04	Removal of Aqueous Uranium by Indigenous Bacteria Under Conditions Relevant for Deep Geological Repository of Nuclear Waste	160
	Dawoon Jeong, Min Hoon Baik, Euo Chang Jung, Ji-Hun Ryu [KAERI], Myoung-Soo Ko [Kangwon Natl. Univ.], Wooyong Um [POSTECH]	
3-P-05	Scenario Development Methods for Deep Geological Repository: (2) Case Study on Disposal Facility for the Spent Nuclear Fuel	161
	Kibeom Son, Karyoung Choi, Jaehyeon Yang, Haeram Jeong, Donghoon Baek, Hyungdae Kim, Kunok Chang, Gyunyoung Heo [Kyung Hee Univ.]	
3-P-06	Digital Analysis Technique for Aperture and Contact Area of Fractures	162
	Yong-Ki Lee, Chae-Soon Choi, Kyung-Woo Park [KAERI]	

3-P-07	Scenario Development Methods for Deep Geological Repository: (1) Research Status and Trends	163
	Karyoung Choi, Kibeom Son, Jaehyeon Yang, Haeram Jeong, Donghoon Baek, Hyungdae Kim, Kunok Chang, Gyunyoung Heo [Kyung Hee Univ.]	
3-P-08	Preliminary Study on Accelerated Life Test of Piezoelectric Sensor	164
	Hyun-Joong Hwang, Changhee Park, Gye-Chun Cho [KAIST], Chang-Ho Hong [KAERI]	
3-P-09	Evaluation of Fracture Filling Minerals and Aperture Distribution of Fractured Granite	165
	Bong-Ju Kim, Ki-Jong Jang, Chung-kyun Park, Jang-Soon Kwon [KAERI]	
3-P-10	Comparison of Two Disposal Concepts for CANDU Spent Nuclear Fuels in Korea	166
	Heui-Joo Choi, Jong Youl Lee, Dong Keun Cho, Jung-Woo Kim [KAERI]	
3-P-11	Analysis and Design of Down Ender and Dismantler for Spent Fuel Assembly Dismantling in Head-End	167
	Younghwan Kim, SunSeok Hong, Seckmin Hong, JuHo Lee, JaeWon Lee, ChangHwa Lee [KAERI]	
3-P-12	A Natural Analogue Study Using Uranium Deposits: Characterization of a Study Site in the Ogcheon Metamorphic Belt	168
	Min-Hoon Baik, Dawoon Jeong, Yeojin Ju, Euo Chang Jung, Ji-Hun Ryu [KAERI]	
3-P-13	Vitrification of Uranium Oxide in Spent Nuclear Fuel Using Iron Phosphate Glass	169
	Cheong Won Lee [KAERI], Jong Heo [POSTECH]	
3-P-14	Development of FEP & Scenario Integrated Information Processing Platform for Long-Term Evolution of Spent Fuel Disposal System	170
	Sanghun Jeon, Sunggyun Jang, Jeong Hyun Lee, Jun Ki An, Jin Ryu, Si On Kim, Won-Seok Kim [NCsquare Co.]	
3-P-15	Two-Phase Flow Analysis in the Full-Scale Emplacement Experiment at Mont Terri Underground Rock Laboratory	171
	Taehyun Kim, Changsoo Lee, Jin-Seop Kim [KAERI], Chan-Hee Park [KIGAM]	
3-P-16	Geochemical Behavior of Cesium and Cobalt in a Near-Surface and Deep Geological Disposal Environments	172
	Jae-Kwang Lee, Jongtae Jeong, Min-Hoon Baik [KAERI]	
3-P-17	Establish Regulatory System and Case Study Database for Deep Geological Disposal System	173
	Jeong Hyun Lee, Sunggyun Jang, Sang Hun Jeon, Gun Hee Kim, Ji Eun Shin, Si On Kim, Won-Seok Kim [NCsquare Co.]	

3-P-18	A Study on the Thermal Conductivity of Bentonite–Sand Mixed Backfill According to Size and Ratio of Sand for Deep Geological Disposal	174
	Yohan Cha, Jae Owan Lee, Mihye Kong, Jin–Seop Kim [KAERI]	
3-P-19	Experimental Study on Piping Erosion in Compacted Bentonite Buffers Considering Buffer–Rock Interactions	175
	Minhyeong Lee, Chang–Ho Hong, Ji–Won Kim, Jin–Seop Kim [KAERI]	
3-P-20	Investigation of Granitic Groundwater Colloids in KURT System by Using Various Analytical Methods	176
	Jun–Hyuk Ha, Jun–Yeop Lee [Pusan Natl. Univ.]	
3-P-21	Experimental Study on Frictional–Healing Behavior of a Rock Joint	177
	Seungbeom Choi, Yongki Lee, Taehyun Kim [KAERI]	
3-P-22	Factors of Geological Risk Assessment for Deep Underground Tunneling in South Korea	178
	Myeong–Hyeok Ihm [Daejeon Univ.]	
3-P-23	Evaluation for Compaction Curves of Compacted Bentonite–Sand Mixtures	179
	Deuk–Hwan Lee, Seok Yoon, Jin–Seop Kim, Ji–Won Kim, Gi–Jun Lee [KAERI]	
3-P-24	Assessment of Evapotranspiration Around KAERI Underground Research Tunnel	180
	Byeong–Hak Park, Eunhye Kwon, Kyung–Woo Park [KAERI]	
3-P-25	Microbial Effects on the Corrosion of Metals in Compacted Bentonite Blocks ..	181
	Seung Yeop Lee, Jae–Kwang Lee, Sang–Ho Lee, Jin–Seok Kim, Jang–Soon Kwon [KAERI]	
3-P-26	Modeling of Uranium Migration Near High–Level Radioactive Waste Repository Using COMSOL–PHREEQC Coupled Model	182
	Won Woo Yonn, Weon Shik Han [Yonsei Univ.], Jeonghwan Hwang, Jung–Woo Kim [KAERI]	
3-P-27	Development of a Numerical Module to Convert DFN to ECPM	183
	Jung–Woo Kim, Yong–Min Kim, Jaewon Lee, Sung–Hoon Ji [KAERI]	
3-P-28	An Efficient Thermal Management Method of Decay Heat in a Disposal Container for Deep Geological Disposal	184
	Jongyoul Lee, Heuijoo Choi, Dongkeun Cho [KAERI]	
3-P-29	Acoustic Emission Characteristics for Bending Failure of Engineered Barrier	185
	Jong–Won Lee, Tae–Min Oh [Pusan Natl. Univ.], Hyunwoo Kim, Min–Jun Kim [KIGAM]	

3-P-30	Bentonite Swelling and Erosion in an Artificial Fracture	186
	Ji-Won Kim, Chang-Ho Hong, Minhyeong Lee, Jin-Seop Kim [KAERI], Song-Hun Chong [Sunchon Natl. Univ.]	
3-P-31	Development and Field Applications of Deep Borehole Hydraulic Testing System (DHTS) Based on Integrated Type Main Frame	187
	Seongho Bae, Hagsoo Kim [GeoGeny Consultants Group Inc.], Eui-Seob Park, Dae-Sung Cheon [KIGAM], Ki-Bok Min [Seoul Natl. Univ.]	
3-P-32	Revisiting A Counter Rotating Plasma Centrifuge For Separating Nuclear Waste	188
	Jun-Ho Seo [JeonBuk Natl. Univ.]	
3-P-33	Comparative Mapping of NEA International and Project-Specific FEP Lists for Deep Geological Disposal of Spent Nuclear Fuel	189
	Jaehyeon Yang, Haeram Jeong, Kibeom Son, Donghoon Baek, Karyoung Choi, Gyunyoung Heo, Kunok Chang, Hyungdae Kim [Kyung Hee Univ.]	
3-P-34	Conceptual Operation System Design and Procedure for Deep Borehole Disposal Using the Wireline Emplacement Method	190
	Hyun Woo Song, Moonoh Kim [KONES Corp.], Dong-Keun Cho, Changsoo Lee [KAERI]	
3-P-35	Development of Copper Alloys for Deep Geological Disposal Canisters of High-Level Radioactive Waste	191
	Minkyu Ahn, Jinwoo Park, Sangeun Kim, Chansun Shin [Myongji Univ.], Dongkeun Cho [KAERI]	
3-P-36	Implementation of Specific Activity Models for C-14 in Terrestrial Environment of a Disposal Facility Using GoldSim	192
	Dongki Kim, Joowan Park [RADCORE], Seo-Yeon Cho, Mi-Seon Jeong [KORAD]	
3-P-37	A Study on the Technology Level of High-Level Radioactive Waste Management: Focused on Delphi and Focus Group Interview Survey Method	193
	Hyeon Woo Oh, Jin Beak Park, Rinah Kim, Minkyu KIM [KORAD], Junghwa Yang [KOMIR]	
3-P-38	Evaluation of KJ-II Bentonite Properties Reacted With Highly Concentrated K ⁺ (1 M) Solutions With Different pHs at 150°C	194
	Ji Hoon Lee, Ho Young Jo [Korea Univ.], Jang Soon Kwon [KAERI]	
3-P-39	Physical and Chemical Properties of Steam-Treated Bentonite	195
	Hong Jae Kim, Ho Young Jo [Korea Univ.]	
3-P-40	Modeling the Effects of Excavation Damaged Zone on the Radionuclide Transport	196
	Jaewon Lee, Jung-Woo Kim [KAERI]	

3-P-41	Mineralogical and Physicochemical Characteristics of Bentonite in the Naah Bentonite Mine, Gyeongju: Natural Analogue Study of the Bentonite Barrier	197
	Yujin Byun, Chanyoung Seo, Ho Young Jo, Young Jae Lee [Korea Univ.], Ji-Hun Ryu [KAERI], Hye-Yoon Cheong, Seong-Wan Park [Clariant (Korea) LTD.]	
3-P-42	Potential Backfill Materials for Deposition Tunnels in a High-Level Waste Repository	198
	Jae Owan Lee, Yohan Cha, Jin-Seop Kim [KAERI]	
3-P-43	Montmorillonite Alteration in Alkaline and Saline Solutions and Its Influences on Retention of Cesium and Iodide by Bentonite Buffer	199
	Ja-Young Goo, Bong-Ju Kim, Jang-Soon Kwon [KAERI], Ho Young Jo [Korea Univ.]	
3-P-44	How to Figure out Aperture Distribution in a Fractured Rock	200
	Chung-Kyun Park, Bong-Ju Kim, Ki-Jong Jang, Jang-Soon Kwon [KAERI]	
3-P-45	Source Term Evaluation Using ORIGEN-S and TRITON for Spent Nuclear Fuel Disposal	201
	Seo-Yeon Cho, Hyungju Yun, Mi-Seon Jeong [KORAD]	
3-P-46	Evaluation of Structural Integrity of Disposal Container Emplacement and Retrieval Device Under Lifting Condition	202
	Young-Oh Lee, Moon-Oh Kim [KONES Corp.], Dong-Keun Cho, Chang-Soo Lee [KAERI]	
3-P-47	Failure Mode of Piezoelectric Sensor for Accelerated Life Test	203
	Changhee Park, Hyun-Joong Hwang, Gyecheon Cho [KAIST], Chang-Ho Hong [KAERI]	
3-P-48	Low-pH Concrete Design for Deep Geological Repository	204
	Seok Gu Gang, Jongwon Jung [Chungbuk Natl. Univ.]	
3-P-49	Design of Data Structure and User Interface of Data Management Platform for FEP and Scenario	205
	In-Young Kim, Jongtae Jeong, Jung-Woo Kim [KAERI]	
3-P-50	Evaluation of Irreversible Cs Adsorption Capacity of Granite Origin Soil	206
	Ayeong Lee, Chul-Min Chon, Jae Gon Kim, Jungho Ryu [KIGAM], Yeongkyoo Kim [Kyungpook Natl. Univ.]	
3-P-51	Fractal Constant Evaluation From Acoustic Signals for Repository Monitoring ..	207
	Chang-Ho Hong, Ji-Won Kim, Jin-Seop Kim [KAERI]	
3-P-52	How to Quantify Well the Pumping Rate From the Aquifer Around the Repository: An Approach	208
	Youn-Myoung Lee, Dong-Keun Cho, Sung-Hoon Ji [KAERI]	

3-P-53	Case Study on In-situ Inspection and Management System for Excavation-Damaged Zone Around the Disposal Caverns in a Spent Nuclear Fuel Repository	209
	Hyung-Mok Kim, Myung Jin Nam, Hanna Jang, Inseok Joung [Sejong Univ.], Eui-Seob Park, Byung-Gon Chae [KIGAM]	

Withdrawal

4분과 중저준위폐기물관리 [ORAL]

4-0-01	The Effect of Isosaccharinic Acid (ISA) on the Solubility of Rhenium(IV) Oxide	213
	Kyungwon Kim, Wooyong Um [POSTECH]	
4-0-02	A Study on the Treatment of Boron-Containing Discharge Water From Nuclear Power Plant Using 1 m ³ /hr Scale of SD-ELIX System	214
	Seongjoo Kang, Seonggeon Kim, Sangwook Lee, Jiyang So, Ducheol Kim, Minho Kwak, Seungil Kim [Elim-Global]	
4-0-03	Review of the Radioactive Waste Acceptance Criteria of the Spent Filter Cartridge Both Home and Abroad for Its Improvement	215
	Suk Nam Lim, Hyun Woo Song, Moonoh Kim [KONES Corp.], Jisoo Yoon, Ji-Hoon Lee [KHNP-CRI]	
4-0-04	Development of High Performance Cement Materials and Solidification Technology for Immobilization of Liquid Radioactive Wastes	216
	Gyeong Il Moon, Seong Ho Yoo, Ho Yeon Yang [Korea Environment Testing and Analysis Laboratory], Dong Hoon Oh [Jung Seok Chemical Corporation]	
4-0-05	A Study on Disposal Suitability of Hanul Glass Fiber Packaging Bag	217
	Na Rae Lee, Moonoh Kim [KONES Corp.], Sang June Park, Sung-Jun Kim, Suil Bang [KHNP]	
4-0-06	A Study on the Carbon Dioxide Adsorption Characteristics Using Radioactive Waste Soil/Waste Concrete	218
	Sangwook Lee, Seongjoo Kang, Duchul Kim, Jiyang So, Seonggeon Kim, Minho Kwak, Seungil Kim [Elim-Global]	
4-0-07	Preliminary Safety Assessment of Worker of Glass Fiber Waste Under Accident Conditions at Disposal Facility	219
	Woo-Yong Kim, Moonoh Kim [KONES Corp.], Sang June Park, Sungjun Kim, Su-il Bang [KHNP]	
4-0-08	Fabrication of Cs Pollucite Ceramic Waste Form and Long-term Leaching Test for Immobilization of Radioactive Cesium	220
	Ga Yeong Kim, Jae Hwan Yang [Chungnam Natl. Univ.], Ki Rak Lee, Jung Hoon Choi, Hwan Seo Park [KAERI]	

4분과 중저준위폐기물관리 [POSTER]

4-P-01	Study on Colloidal Impact of Glass Fiber Particle Considering Sorption Ability	223
	Hyun Woo Song, Moonoh Kim [KONES Corp.], Sang June Park, Sungjun Kim, Su-il Bang [KHNP]	
4-P-02	Evaluation of Leaching Behavior and Thermal Stability of Silver Tellurite Glass ..	224
	Seong Sik Shin [Chungnam Natl. Univ.], Hyun Woo Kang, Jung-Hoon Choi, Ki Rak Lee, Hwan-Seo Park [KAERI]	
4-P-03	A Design of Optimal Repackaged Algorithm for Reduction of Low-and Intermediate-level Radioactive Waste	225
	Hee-Seoung Park, Il-Sik Kang, Dong-ju Lee, Choong-Wie Lee, Won-Hyuk Jang, Jong-Jin Kim, Jin-Woo Lee [KAERI]	
4-P-04	Solubility of Radionuclides Under Near and Far Field Conditions: Effect of Organic Complexing Agents	226
	Bolam Kim, Ahsan Abdul Ghani, Youngsu Lim, Dae Sung Lee [Kyungpook Natl. Univ.]	
4-P-05	A Study on Low-Radiation Waste Resin Treatment Using Plasma Torch Melting Technology	227
	Jeongsu Jeong, Seoyong Choi, Sunghoon Hong, Mingyu Cho [KHNP-CRI]	
4-P-06	A Parametric Study on Shadow Zone Effect in Near-Field Radionuclide Transport Modeling for an Engineered Vault Type LILW Disposal Facility ..	228
	Wontak Lee, Dongki Kim, Joowan Park [RADCORE]	
4-P-07	Development Trend of Radwaste Characterization Technology for Safe Disposal	229
	Seok-Ju Hwang, YoungHwan Hwang, Cheon-Woo Kim [KHNP-CRI]	
4-P-08	Degradation of Cellulosic Materials Into Isosaccharinic Acid Both Under Partially Aerobic and Completely Anaerobic Conditions	230
	Bolam Kim, Sanghwa Oh, Dae Sung Lee [Kyungpook Natl. Univ.], Jiseon Jang, Seho Choi, Myunggoo Kang, Jaechul Ha, Chun-Hyung Cho [KORAD]	
4-P-09	Air Conditioning Condensate Contamination From Radiation Management Area of the DU Facility	231
	Youngsang Lee, Youngwoo Kwon, Seonho Noh [KAERI]	
4-P-10	Analysis of Methodology for Deriving the Total Amount of SKB Cellulose-Containing Radioactive Waste	232
	Bora Yang, Jaechul Ha, Myunggoo Kang, Seho Choi [KORAD]	
4-P-11	Optimization of Reactor Vessel Internals Segmentation of Kori Unit 1	233
	Kyu-Sang Hwang, Chang-Lak Kim [KINGS]	

4-P-12	^{239}Pu and ^{99}Tc Sorption to the Cementitious and Natural Rock Barriers: Effect of Organic Complexing Agents (EDTA and ISA)	234
	Seongju Lee, Youngsu Lim, Bolam Kim, Daesung Lee [Kyungpook Natl. Univ.]	
4-P-13	Cesium Adsorption on Engineered and Natural Barriers in a Fixed-bed Column System: Effects of EDTA and ISA	235
	Juhui Park, Youngsu Lim, Bolam Kim, Dae Sung Lee [Kyungpook Natl. Univ.]	
4-P-14	Effect of Organic Complexing Agents on Sorption and Mobility of Radionuclide ^{90}Sr by Engineered and Natural Rock Barriers at L/ILW Disposal Facility	236
	Beomgeol Jang, Ahsan Abdul Ghani, Bolam Kim, Youngsu Lim, Dae Sung Lee [Kyungpook Natl. Univ.]	
4-P-15	Effect of Chelating Agents and Cement Degradation States on the Transport Behavior of ^{238}U in Engineered and Natural Barriers	237
	Youngsu Lim, Juhui Park, Dae Sung Lee [Kyungpook Natl. Univ.]	
4-P-16	Self-disposal of Metal Radioactive Waste	238
	Kyeong Min Park, Si Hyung Kim, Jong Hun Kim, Yun Jeong Hong, Seong Jae Cheon, Sunghwan Kim, Yong Jin Jeong [KAERI]	
4-P-17	Tellurite Glasses for Immobilization of Selenium	239
	Jae-Young Pyo, Jung Hoon Choi, Hyun Woo Kang, Ki Rak Lee, Hwan Seo Park [KAERI]	
4-P-18	A Study on Suitability for the Vitrification of Radioactive Asbestos Waste in Kori Unit1 Decommissioning	240
	Hwa-jun Yeom, Chang-Lak Kim, Chang-Sig Kong [KINGS], In-Sun Jang [Dawonsys]	
4-P-19	Reduction and Recycling of Nuclear Dismantled Concrete Waste	241
	Jin Ryu, Jun Ki An, Gun Hee Kim, Ji Eun Shin, Won-Seok Kim [NCsquare Co.]	
4-P-20	International Case Study on Economic Estimation for Thermal Plasma- Gasification	242
	InJun Lee, HyungJun Kim [KEPCO KPS], Seoyong Choi [KHNP-CRI]	
4-P-21	Removal of Uranium Ions From Sulfate-Based Acidic Soil Washing Effluent Using Ion-Exchange Resins	243
	Hyun-Kyu Lee, Seeun Chang, Wooshin Park, Sunghbin Park, Hongrae Jeon [KAERI]	
4-P-22	Introduction on the 3.5th Generation Plasma Torch Melting Facility	244
	Seoyong Choi, Jeongsu Jeong, Sunghoon Hong [KHNP-CRI]	
4-P-23	Clearance Level Radioactive Waste Treatment for Reuse	245
	YuJeong Choi, Yoondo Oh, Tack-Jin Kim [KAERI]	

Withdrawal

4-P-24	A Study on the Development of the Evaluation Module for Decommissioning of Heavy Water Reactors	246
	Hyun Young Shin, Kyu Tae Park, Sung Ryul Kim, Jung Min Oh, Ba Ro Lee, Ji Ung Kim, Hye Jin Kim, So Yun Jeong, Wook Jae Yoo [Orbitech]	
4-P-25	Analysis on US NRC Technical Position for Concentration Averaging of Radioactive Waste	247
	Jiyoung Yi, Chang-Lak Kim [KINGS]	
4-P-26	Study on Recycling Strong Acidic Wastewater Containing a Radionuclide for the Radioactive Waste Reduction	248
	Hee-Chul Eun, Na-On Chang, Tack-Jin Kim, Woo-Shin Park [KAERI]	
4-P-27	A 100 kW Class Transferred Type Plasma Torch for Radioactive Waste Melting Applications	249
	Darian Figuera-Michal, Dong-Hyun Lee, Nam-Ki Lee, Shi-Young Yang, Jun-Ho Seo [JeonBuk Natl. Univ.]	
4-P-28	A Conceptual Study on Radioactive Waste Solidification Treatment Using Plasma Torch Melter	250
	Jihyun Kim, HyungJun Kim [KEPCO KPS], Seoyong Choi [KHNP-CRI]	
4-P-29	Development of Procedures for Compression and Packaging of Spent Filter	251
	Narae Lee, Moonoh Kim [KONES Corp.], Ji Soo Yoon, Ji Hoon Lee [KHNP-CRI]	
4-P-30	Performance Test of A 100 kW Class Mobile Plasma Melting System for Non-combustible Wastes	252
	Darian Figuera-Michal, Dong-Hyun Lee, Nam-Ki Lee, Jun-Ho Seo [JeonBuk Natl. Univ.], Sang-Min Jeong [ORION EnC]	
4-P-31	Evaluation of the Mobility of Glass Fibers in the Environment of the Gyeongju Radioactive Waste Repository	253
	Sang June Park, Hang-Rae Cho, Suil Bang [KHNP], Moonoh Kim, Hyun Woo Song [KONES Corp.]	
4-P-32	Proposal for the Composition of the Waste Profile (Draft) for Large Package of Decommissioning Waste	254
	Hoseog Dho, Hyungoo Kang, Yeseul Cho, Chunhyung Cho [KORAD]	
4-P-33	Design of Three Phase AC (Alternative Current) Arc Plasma System For Treatment of Radioactive Slurry Wastes	255
	Dong-Hyun Lee, Darian Figuera-Michal, Hae-Won Park, Sang-Yun Baek, Jun-Ho Seo [JeonBuk Natl. Univ.]	

4-P-34	Evaluation of Seawater Intrusion Characteristics Using the Long-range GPR Systems for Coastal Line at the Site of LILW Disposal Facilities	256
	Seung Gyu Hyun, Hoseon Choi [KINS], Youngchul Yu [KOTAM]	
4-P-35	The Evaluation of Tritium Behavior by Replacing the Reflector Ion Exchanger Resin With HANARO	257
	Hoon-Jo Cho, Mun-Jo Choi, Kwang-Young Lee, Hee-Gon Kim, Ho-Young Choi [KAERI]	
4-P-36	Characteristics of Waste-form Containing Chemical Decontamination Waste Immobilized by Low-temperature Sintering	258
	Naon Chang, Hee-Chul Eun, Won-Hyuk Jang [KAERI]	
4-P-37	The Effect of ISA on the Sorption and the Solubility of Nickel	259
	Soojeong Ji, Wooyong Um [POSTECH]	
4-P-38	Simulation of Liquid Waste From Decontamination of Small- and Medium-sized Metal Wastes of NPP Decommissioning and Design Conditioning and Treatment Process	260
	Seong Ho Yoo, Ki Seock Cho, Kwang Nam Kim, Ho Yeon Yang [Korea Environment Testing and Analysis Laboratory]	
4-P-39	Considerations for Designing Solidification Devices of Dispersible Radioactive Waste	261
	Dong-Sik Jin, Yong-Ho Hong, Hui-Gyeong Kim, Sang-Soo Kwak, Jae-Geun Lee [Nuclear Environment Technology Co., Ltd.]	
4-P-40	A Conceptual Design for the Development of a Gripper Device for Lifting Work of Polymer Concrete High Integrity Container (PC-HIC) Packages ..	262
	Hang-Rae Cho, Hyung-Jin Roh, Sang-June Park [KHNP], Mi-Hyun Lee [KHNP-CRI]	
4-P-41	Study on Coincidence Summing Corrections Using LVis Software in Analysis of the Low-level Radioactive Waste	263
	Kyunghun Jung, Minyoung Jung, Jung Bo Yoo, Tae-Hong Park [KAERI]	
4-P-42	Solubility Experiment for the Potential Role of Concrete Admixture on the Radionuclide Mobility	264
	Donghun Pak, Jueun Kim, Kyungwon Kim, Wooyong Um [POSTECH]	
4-P-43	Release of Sr From Sr-Coprecipitated Calcite Under Various Environment Conditions	265
	Nara Jeong, Wooyong Um [POSTECH]	
4-P-44	Sorption of Organic Complexing Agents to Engineering Barriers of a Radioactive Waste Disposal Site	266
	Jinseob Kim, Ahsan Abdul Ghani, Bolam Kim, Youngsu Lim, Dae Sung Lee [Kyungpook Natl. Univ.]	

4-P-45	Safety Tests of Waste Containers for Packaging of NPP Decommissioning Wastes	267
	Ju-Chan Lee, Jongmin Lim, Yun Young Yang, Sangsoon Cho, Jeong-Yong Park [KAERI]	
4-P-46	Review on Measurement Method of Level Using Differential Pressure in Nuclear Facility	268
	Yoon-do Oh, Hyun-myung Jung, Dae-seok Hong, Dong-han Yoo, Hyun-kyu Lee, Tack-jin Kim [KAERI]	
4-P-47	Process for Regulatory Clearance of Wood Waste at KAERI	269
	Han Byeol Kang, Jong Kwang Lee, Tack-Jin Kim [KAERI]	
4-P-48	Calculation of Radiation Dose Using Computational Program NRCDOSE	270
	Han Byeol Kang, Ki-Baek Shin, Tack-Jin Kim [KAERI]	
4-P-49	Development of Integrated Decontamination Equipment to Remove Concrete Surface Contamination	271
	Byeong-Ho Kim, Ki-Hyun Kwon, Hyeon-Oh Park [Sunkwang T&S], Byung-Seon Choi [KAERI]	
4-P-50	Upcycling of Radioactive Wastes to Foam Filter for Performance Improvement Using Sacrificial Material	272
	Jang-Yul Kim, Yong-Soo Jung, Gangho Lee [Research Institute of Industrial Science & Technology]	
4-P-51	Concept Designs of a Sampler Device for Taking Representative Samples From Waste Packages of High Integrity Container (HICs)	273
	Hang-Rae Cho, Sang-June Park, Su-Il Bang [KHNP], Young-Hwan Hwang [KHNP-CRI]	
4-P-52	The Refinement System Development for Lubricant Oil Waste in the Form of Organic Compounds Contaminated With Radioactive Materials	274
	Hui-Gyeong Kim, Yong-Ho Hong, Dong-Sik Jin, Sang-Soo Kwak, Dong-Cheol Kim [Nuclear Environment Technology Co., Ltd.]	
4-P-53	Automatic Waste Position Location and Its Information Administration for Safe Disposal Site Management Using Beacon	275
	Sungjong Kim, Kweonwoo Sohn, Changhwan Cho [Mirae-en]	
4-P-54	A Safety Assessment of a Radioactive Waste Disposal Facility: A Well Scenario	276
	Mi-Seon Jeong, Dongkook Park, Soo-Gin Kim, Jae-Yeol Cheong [KORAD]	
4-P-55	Preliminary Modeling to Characterize the Impact of Surrounding Structures on Neutron Activation in Unshielded Cyclotron Facility Using MCNP With FISPACT	277
	Hyun Seok Song, Jae Hak Cheong [Kyung Hee Univ.]	

4-P-56	Status Update for Contaminated Water Management in Fukushima Daiichi NPP	278
	Sunghoon Hong, Seok-Ju Hwang, Cheon-Woo Kim [KHNP-CRI]	
4-P-57	Review of New Regulatory Guidelines in Relation to the Off-site Dose Calculation	279
	Deuk-Man Kim, Seok-Ju Hwang, Cheon-Woo Kim [KHNP-CRI]	
4-P-58	Preliminary Consideration of Radiological Risk Associated With Transport of Spent Fuel and Radioactive Waste in Environmental Impact Assessment for Korean NPPs	280
	Younghoon Hwang, Jae Hak Cheong [Kyung Hee Univ.]	
4-P-59	An Overview of the Revision History of Assessment Methodology of Radioactive Source Term for Normal Operation of Light Water Reactors	281
	Ji-Young Song, Kwan-Hee Lee [KINS]	
4-P-60	Trends of ^3H Decontamination Technology and Design of Pilot Hybrid Technology Facilities for ^3H Contaminated Water Treatment	282
	Jiyang So, Doocheol Kim, Seonggeon Kim, Sangwook Lee, Seongjoo Kang, Minho Kwak, Seungil Kim [Elim-Global]	
4-P-61	Review of Licensing Process of the 2 nd Phase Low Level Radioactive Waste Disposal Facility in Gyeongju	283
	Yohan Kim, Eunjin Seo, Jinyong Park, Chan Woo Jeong, Sangmyeon Ahn, Jeheon Bang, Jungjin Kim, Sang-Ho Lee, Jinmo Ahn, Sangsu Park [KINS]	
4-P-62	Study on Test Requirements Applied to Immobilized Solid Form	284
	Jisoo Yoon, Ji-Hoon Lee [KHNP-CRI]	
4-P-63	Benefit Analysis for Melting of Metallic Intermediate Level Waste	285
	Moonoh Kim [KONES Corp.], Yoonji Choi, Jeonghyun Yoon [KORAD]	
4-P-64	A Study on Predition of Annual Disposal Capacity for Gyeongju Radioactive Waste Repository Using System Dynamics Modeling	286
	Yuwhwa Son, Jachul Ha, Chunhyung Cho, Hyunchul Kwon [KORAD]	
4-P-65	A Case Study of Clearance of Volumetric Contaminated Wood Wastes	287
	Yunjeong Hong, WonHyuk Jang [KAERI]	
4-P-66	Examination for Repackaging Method of Intermediate Level Waste	288
	Moonoh Kim [KONES Corp.], Yoonji Choi, Jeonghyun Yoon [KORAD]	
4-P-67	Corrected Off-Gas Illustrative Model for 6 MW Molten Salt Reactor	289
	Ju Ho Lee, Seok-Min Hong, ChangHwa Lee [KAERI]	

4-P-68	The Characteristics of Solidified Sludge According to the Distribution of Ferro Frit Powder in Sludge	290
	Ki Joon Kang, Sia Hwang, Hee Reyoung Kim [UNIST]	

5분과 제염해체 [ORAL]

5-0-01	Development of NPP Cost Estimation Program Through 3D and Unit Cost Factor (UCF)	293
	Minhee Kim, Chang-Lak Kim, Sanghwa Shin [KINGS]	
5-0-02	Development of a Platform That Enables Generation of Cutting Paths and Process Simulation for Decommissioning of Reactor Vessel Internals (RVI) Within a Virtual Environment	294
	Jae-Seok Ki, Kyo-Chan Hwang, Ju-Ho Choi [Sangmyung Univ.], Jeong Suh, Ryoan-han Kim [KIMM]	
5-0-03	Development of 3D Imaging Equipment Available in Radioactive and Underwater Environments	295
	Dongjun Hyun, Sungmoon Joo, Ikjune Kim, Jaehyun Ha, Jonghwan Lee [KAERI]	
5-0-04	Development of Integrated Decontamination Process Using Supercritical Carbon Dioxide and Ethanol for Soil Contaminated With Cs, Sr, U	296
	Seungil Ha, Kwangheon Park, Jinhyun Sung [Kyung Hee Univ.]	
5-0-05	Analysis of the Necessity of Building New Vessel for the Transport of Decommissioning Radioactive Waste	297
	Sanghyeon Eom, Moonoh Kim, Woo Yong Kim [KONES Corp.]	
5-0-06	A Study on Solidification of Uranium Contaminated Soil Waste Based on Glass-ceramization	298
	Jaewoong Hwang, Jaseung Koo [Chungnam Natl. Univ.], Keunyoung Lee [KAERI]	
5-0-07	A Study of the Physical and Chemical Properties of Aerosols Generated by Laser Metal Cutting	299
	Hyun Jin Boo, Hwa Jeong Han, Jonghyeon Kim, Su Hyeon Lee, Su Hyeon Kim, Byung Gi Park [Soonchunhyang Univ.], Min-Ho Lee, Hee Kwon Ku [FNC Technology], Dae-Won Cho [KIMM]	

5분과 제염해체 [POSTER]

- 5-P-01 Preliminary Mass Flow Diagram of a Thermochemical Process for Decontamination of Radioactive-combustible Waste 303
Min Ku Jeon, Sung-Wook Kim, Maengkyo Oh, Hee-Chul Yang, Keunyoung Lee [KAERI]
- 5-P-02 Safety Study on Residual Power Transmitted From Laser-cut Object for Nuclear Decommissioning Application 304
Jae Sung Shin, Ki-Hee Song, Seong Y. Oh, Seung-Kyu Park [KAERI]
- 5-P-03 Control Measures of Safety and Risks for Induction Melting Facilities of Radioactive Metal Waste 305
Kwan Seong Jeong, In-Ho Yoon, Keun Young Lee [KAERI], Jung Jun Lee [KINS]
- 5-P-04 Development of 3D Mapping Procedure for Derived Reasonable Work Plan in the Case of Decommissioning Nuclear Power Plant 306
Jaeryang Kim, Sangbok Lee, Dongyoon Kim, Jungmook Kim, Chiil Ohk, Giwon Lee, Doohee Lee, Jungki Kang [LC Gen Co. Ltd]
- 5-P-05 The Establishment Strategy of Effective Management Plan for Stakeholder Engagement During Decommissioning of Nuclear Facilities 307
Kwan Seong Jeong, In-Ho Yoon, Chang Hyun Roh [KAERI]
- 5-P-06 Understanding of Scintillators for Radiation Detection Under Intense Gamma-rays Fields for Nuclear Decommissioning 308
Changhyun Roh, Su Jung Min, Seon Kwang Yoon, Chae Hun Lee, Sang Yoon Park, Sang Bum Hong, Bum Kyung Seo [KAERI]
- 5-P-07 Magnetic Hierarchical Titanium Ferrocyanide for the Selective Removal of Radioactive Cesium From Water 309
Hee-Man Yang [KAERI]
- 5-P-08 Surface Decontaminants Composed of Poly (Vinyl Alcohol) and Phenylboronic Acid-grafted Poly (Methyl Vinyl Ether-alt-Mono-Sodium Maleate) 310
Hee-Man Yang, Sung-Wook Kim, Jong-Yun Kim, Hyung-Ju Kim [KAERI]
- 5-P-09 Management Prospects of Solid Radioactive Waste Generated From Fukushima Nuclear Power Plant Accident 311
Seok-Ju Hwang, Sunghoon Hong, Cheon-Woo Kim [KHNP-CRI]
- 5-P-10 Improving the Detection Efficiency of the Radiation Detector Through Secondary Solute Modification 312
Su Jung Min, Jae Hak Cheong [Kyung Hee Univ.],
Yong Dae Park, Bum Kyung Seo, Chang Hyun Roh, Sang Bum Hong [KAERI]

5-P-11	Development of Remote Machining Technique in Hotcell for Compact Tension Specimens of Materials Harvested From Nuclear Reactor	313
	Kisoo Heo, Younggwan Jin, Seungje Baik, Minchul Kim [KAERI]	
5-P-12	Method of Preventing Radioactive Leakage of Exhaust Gas Treatment Facility ...	314
	In-Jun-Lee, Ju-Sakong [KEPCO-KAP], Seoyoung-Oh [KHNP-CRI]	
5-P-13	Inquiry Into the Method to Reduce the Amount of Radioactive Waste	315
	Injun-Lee, Panjae-Kim, Sangjun-Kim [KEPCO-KAP], Taeyoung-Choi [KHNP-CRI]	
5-P-14	Comparison of ICRP Publication Dose Conversion Factors and Estimation of Exposure Dose for Decommissioning Workers	316
	Jin Ho Son, Seong Jun Kim, Wo Suk Choi, Hwa Pyoung Kim, Chang Ju Song, Hee Jin Kim, Tae Young Kong [Chosun Univ.]	
5-P-15	Reduction Process of Transferred Arc Plasma Melting System	317
	Ki-Man Park [VITZRO NEXTECH]	
5-P-16	Development of Plasma Melting System for the Disposal of Composite Materials	318
	Byung-Soo Yoo [VITZRO NEXTECH]	
5-P-17	Development of Scan Simulator for Automation of Nuclear Facility 3D Modeling	319
	Sungmoon Joo [KAERI]	
5-P-18	An Approach to Enhancement of Underwater Laser Cutting Process Monitoring Video	320
	Sungmoon Joo, Dongjun Hyun, Ikjune Kim, Jaehyun Ha, Jonghwan Lee, Taeyoung Ko [KAERI]	
5-P-19	Analysis of the Foreign Cases of Building Reuse Safety Assessment After Decommissioning of Nuclear Facilities	321
	Min Seong Kim, Ji Woo Kim, Yong Ho Jin, Kwang Pyo Kim [Kyung Hee Univ.]	
5-P-20	Application Study of Sequential Soil Flusing for Uranium Removal	322
	Seeun Chang, Hyun-Kyu Kim, Woo Shin Park, Sungbin Park, Hongrae Jeon [KAERI]	
5-P-21	Radiation Management During Full System Decontamination for NPP Decommissioning	323
	Jong-hoa Kim, Ji-hyun Yu, Bung-chae Lee, Jang-soo Suh [SAE-AN Enertech Corp.]	
5-P-22	Considerations for NPP Decommissioning Project Risk Management	324
	Gi-Lim Kim, Hyung-Woo Seo, Sang hyun Lee, Ji-hwan Yu, Hye-In Kim [KHNP-CRI]	
5-P-23	A Case Study on Methodology of Dose Calculation for the Embedded Pipe of Decommissioning NPP	325
	Jihwan Yu, Minchul Kim, Gi-lim Kim [KHNP-CRI]	

5-P-24	A Case Study on the Remediation of Embedded Piping of Trojan Decommissioning NPP	326
	Jihwan Yu, Hyung-woo Seo, Gi-lim Kim, Hyein Kim [KHNP-CRI]	
5-P-25	Study on Thermal Stability and Structural Changes of the Sr-exchanged Natrolite	327
	Junhyuck Im [KAERI], Jaewoo Jung [KIOST], Kiho Yang, Yongmoon Lee [Chonnam Natl. Univ.], Donghoon Seoung [Pusan Natl. Univ.]	
5-P-26	Feasibility Study on High Count Rate Application of LaBr ₃ Detector With a High Speed Digitizer for Nuclear Facility Decommissioning	328
	Chaehun Lee, Seonkwang Yoon, Sang-bum Hong, Bumkyung Seo [KAERI]	
5-P-27	Radioactivity Calculation Considering Plant-specific Operation History for Decommissioning of PWR Nuclear Power Plant	329
	Youngjae Maeng, Kyungsik Kim, Mijuong Lim, Choonsung Yoo [KRIST]	
5-P-28	Proposal of Evaluation Matrix for Clearance Waste Disposal Plan	330
	Kunsu Lim, Chang-Lak Kim, Sanghwa Shin [KINGS]	
5-P-29	A Review of Remediation Actions and Applicable Scenarios to be Considered at the Time of License Termination for NPPs Decommissioning	331
	Hyung-woo Seo, Ji-hwan Yu, Gi-lim Kim, Hyein Kim [KHNP-CRI]	
5-P-30	Evaluation of Dose to Public Due to Hypothetical Events During Dismantling of Steam Generator	332
	Kyeong Ju Lee, Chang Lak Kim [KINGS]	
5-P-31	Computational Model Development of a Combined Electrolysis and Catalytic Exchange (CECE) Process for Heavy Water Pre-enrichment and Detritiation	333
	Kwang-Rag Kim, Jong-Yun Kim, Sang-Ho Lim, Do-Hee Ahn [KAERI]	
5-P-32	Review of Transport System for Decommissioning Waste Containers	334
	Woo-Yong Kim, Young Oh Lee, Tae Su Kim, Moonoh Kim [KONES Corp.]	
5-P-33	A Study of Dismantling Steam Generators for Kori Unit 1, 2	335
	Geunyoung Goh, Chang-Lak Kim [KINGS]	
5-P-34	Dose Evaluation on the Post-closure Scenario of Self-disposal Landfill for Decommissioning Metal/Concrete Wastes	336
	Jaewon Park, Juyub Kim, Hyungi Byun [FNC Technology]	
5-P-35	Case Study of Decommissioning Waste Treatment Method of Zion Nuclear Power Station	337
	Cheong Won Lee [KAERI]	

5-P-36	Activated Carbon Waste Pretreatment System for Radioactive Waste Up-cycling	338
	Kirak Lee, Hwan-Seo Park, Geun-il Park [KAERI], Sang Won Shin, Dong Chul Jang [HANIL Nuclear]	
5-P-37	Paraffin Separation From Paraffin Wasteform Via Evaporation and Thermal Decomposition Method	339
	Kirak Lee, Hyeon Jin Eom, Hwan-Seo Park, Jung-Hoon Choi, Hyun Woo Kang [KAERI]	
5-P-38	Hydrogen Isotope (H, D, T) Separation Technology Development	340
	Chan Woo Park, Ilgook Kim, Hyung-Ju Kim, Chan Hyun Rho, Younho Shin, In-Ho Yoon, Hee-Man Yang [KAERI]	
5-P-39	Conflict Resolution for Each Stakeholder on the Clearance Waste Recycling From the Decommissioning of Nuclear Power Plants	341
	Dong-Jin Kim, Ye-Jeon Lim, So-On Park, Seong-Hyeon Lee, Byung-Sik Lee [Dankook Univ.]	
5-P-40	PERT Chart Based Laser Cutting Process Control on a Digital Simulation Platform	342
	Ikjune Kim, Dongjun Hyun, Jonghwan Lee, Sungmoon Joo, Jaehyun Ha, Taeyoung Ko [KAERI]	
5-P-41	Evaluation of Solubility of Ethanol, 2-propanol in Supercritical Carbon Dioxide for the Application of Soil Decontamination Process	343
	Seungil Ha, Kwangheon Park, Jinhyun Sung [Kyung Hee Univ.]	
5-P-42	Demonstration Test for Chemical Waste Decomposition & Treatment System(CWDS)	344
	Ju-Hyeon Park, Jung-Hyun Lee, Sung-Hyun Lee, Ki-Chul Kim, Chung-kyu Lee [KEPCO KPS]	
5-P-43	~~Screening of Porous Materials for Separation of Hydrogen Isotopes Based on Cryosorption~~	~~345~~
	~~Chan Woo Park, Hyung-Ju Kim, In-Ho Yoon [KAERI]~~	
5-P-44	Changes to the Design of CHEONG JEONG NURI for the Transport of Decommissioning Radioactive Waste	346
	Sanghyeon Eom, Moonoh Kim, Woo Yong Kim [KONES Corp.]	
5-P-45	Scale-up Study for Volume Reduction Process of Radioactive Concrete Waste From Decommissioning Projects	347
	Maengkyo Oh, Chang-Ha Lee [Yonsei Univ.], Keunyoung Lee, Min-Ku Jeon, Richard Ian Foster [KAERI]	
5-P-46	Technology Development for Advanced Volume Reduction of Radioactive Concrete Waste From NPP Decommissioning	348
	Keunyoung Lee, Maengkyo Oh, Richard I. Foster, Kwang-Wook Kim, Sungjune Sohn, Min Ku Jeon, Bum-Kyoung Seo [KAERI]	

Withdrawal

5-P-47	Demonstration Test of Full System Decontamination and Its Operation Review	349
	Sung Hyun Lee, Jung-Hyun Lee, Ju-Hyeon Park, Sung-Sik Yoon, Jeong-Yeop Cha, Seung-Ki Lee, Sung-Su Kim, Chung-Kyu Lee, Ki-Chul Kim [KEPCO KPS]	
5-P-48	A Study on the Sintering Stabilization Method of Cs-contaminated Spent Adsorbent	350
	Keunyoung Lee, Maengkyo Oh, Jaewoong Hwang, Min Ku Jeon [KAERI]	
5-P-49	A Study of Effect of Trace Materials in Activation Analysis	351
	Mi-Hyun Lee, Young Hwan Hwang, Cheon-Woo Kim [KHNP-CRI]	
5-P-50	A Study on Waste Management of ZION Nuclear Power Plant Decommissioning	352
	Young Hwan Hwang, Mi-Hyun Lee, Seok-Ju Hwang [KHNP-CRI]	
5-P-51	Study on Creating and Optimizing a Task Sequence for Robotic Underwater Laser Cutting Using a Digital Manufacturing System	353
	Jaehyun Ha, Dongjun Hyun, Sungmoon Joo, Ikjune Kim, Jonghwan Lee [KAERI]	
5-P-52	Performance Tests on the Plasma Arc Melter-Molten Salt Oxidizer Combination System – 2. Waste Form Integrity of Melting Solidification Products	354
	In Sun Jang, Na Young Kim, Jae Hyuk Choi [Dawonsys], Hee Chul Yang, Hee Chul Eun [KAERI]	
5-P-53	Performance Tests on the Plasma Arc Melter-Molten Salt Oxidizer Combination System – 1. Destruction of Organics and Retention of Metals/Radionuclides ...	355
	In Sun Jang, Na Young Kim, Jae Hyuk Choi [Dawonsys], Hee Chul Yang, Hee Chul Eun, Hyung Ju Kim [KAERI]	
5-P-54	Study on the Decomposition for Radioactive Wastes Containing a Hazardous Material (Asbestos) in a High-Temperature Melt Oxidation System	356
	Na-Young Kim, In-Sun Jang, Jae-Hyuk Choi [Dawonsys], Hee-Chul Yang, Hee-Chul Eun [KAERI]	
5-P-55	Separation of Hydrogen Isotopes by Water Electrolysis	357
	Chan Woo Park, Ilgook Kim [KAERI]	
5-P-56	Conflicts Over the Clearance Waste Recycling From the Decommissioning of Nuclear Power Plants	358
	Ye-Jeon Lim, Dong-Jin Kim, So-On Park, Byung-Sik Lee [Dankook Univ.]	
5-P-57	A Revised Method for Estimating Initial Provision of Nuclear Power Plant (NPP) Decommissioning Cost	359
	Seunghyuk Jung, Hyunwoo Song [KONES Corp.]	
5-P-58	A Preliminary Modeling to Characterize Major Factors Affecting Neutron Activation of Medical Cyclotrons Using MCNP, FISPACT-II and FLUKA	360
	Se Geun Lee, Jae Hak Cheong [Kyung Hee Univ.]	

6분과 방사선환경 및 안전 [ORAL]

- 6-0-01 Review on the Evaluation Methodology of Radioactive Source Term for Normal Operation of NuScale SMR 363
Ji-Young Song, Gyeong-Mi Kim, Su-Ji Yoon, Ho-Jin Lee, Kwan-Hee Lee [KINS]
- 6-0-02 Radiation Dose Assessment of Thorium-containing Gas Mantle for Use and Disposal Scenarios 364
Jimin Shin, Beomseok Ko, Hee Seo [JeonBuk Natl. Univ.],
Meeryeong Kim, Minkyung Kim, Sangmin Lee [KINS]
- 6-0-03 Verification of Monte Carlo Simulation Model on Seafood Monitoring System for Radioactive Material Screening 365
Hyosang Choi, Hee Seo [JeonBuk Natl. Univ.]
- 6-0-04 Radiation Safety Evaluation for Radioactive Waste Processing in Hot Cell Facility 366
Kyung Hun Shin, Hee Reyoung Kim [UNIST]
- 6-0-05 LSC Quenching Correction of H-3 Leached Sample of Paper Packaging Material 367
Changhyun Ryu, Sangyong Park, Junghwan Hong, Jusngsuk Park [KEPCO NF],
Kwangpyo Kim [Kyung Hee Univ.]
- 6-0-06 Shielding Thickness of the Fuel Handling Hot Cell at the Spent Fuel Re-packaging Facility 368
Joon Gi Ahn [KEPCO E&C]
- 6-0-07 Occupational Dose Analysis for Fire Accident During NPP Decommissioning ... 369
Hyunjin Lee, Chang-Lak Kim, Sang-rae Moon, Sun-kee Lee [KINGS]
- 6-0-08 A Study on the Status of Multi-nuclide Removal System (ALPS) and Dilution/Discharge Process on Fukushima NPP Site 370
Myeonggi Cha, Gisu Lee, Min Baek [POSTECH]

6분과 방사선환경 및 안전 [POSTER]

- 6-P-01 The Mapping Program for the Display of Radiation Distribution Using Aerial Radiation Monitoring Data 373
Hwa Jeong Han, Byung Gi Park [Soonchunhyang Univ.],
Sang Hun Shin, Hee Kwon Ku, Min Beom Heo [FNC Technology]

6-P-02	Development and Field Tests of a Mobile Radiation Monitoring System Using Public Transportation	374
	Jin Sik Choi, Jae Wook Kim, Han Young Joo, Jeong Yeon Lee, Joo Hyun Moon [Dankook Univ.]	
6-P-03	Spectroscopy- $H^*(10)$ Rate Conversion Function $G(E)$ for a Plastic Scintillation Detector	375
	Jisung Hwang, Junhyeok Kim, Kilyoung Ko, Wonku Kim, Jaehyun Park, Gyohyeok Song, Sangho Lee, Gyuseong Cho [KAIST]	
6-P-04	Development of Virtual Gamma-Ray Energy Spectrum Simulator Program for Training	376
	Yoomi Choi, Young-Yong Ji, Sungyeop Joung, Eunjoong Lee [KAERI]	
6-P-05	Miniaturization and Applicability Evaluation of Collimating Device for the Self-Attenuation Correction in Low-Energy Gamma-Ray Spectrometry	377
	Jun-Ho Lee, Jong-In Byun [KINS]	
6-P-06	Feasibility Study on Gamma Energy Spectroscopy of Bismuth Nanoparticles-loaded Plastic Scintillator for Portal Monitoring Using Monte Carlo Simulations	378
	Taeseob Lim, Siwon Song, Jae Hyung Park, Jinhong Kim, Seunghyeon Kim, Hyungi Byun, Seokhyeon Jegal, Bongsoo Lee [Chung-Ang Univ.]	
6-P-07	Design Improvement on Reducing Angular Dependence of Gamma Probe for Detecting Radioactive Contamination	379
	Sanggeol Jeong, Giyoon Kim, Jaeyeong Jang, Heejun Chung, Myungsoo Kim [KINAC]	
6-P-08	Minimization for Temperature Dependence of the SiPM Adapting Count Rate Compensation and Operation Sequence	380
	Giyoon Kim, Sanggeol Jeong, Jaeyeong Jang, Heejun Chung, Myungsoo Kim [KINAC]	
6-P-09	Scintillating Detection of Fe-55 for Radioactive Aerosol Monitoring	381
	Jae Hyung Park, Siwon Song, Seunghyeon Kim, Jinhong Kim, Taeseob Lim, Seokhyeon Jegal, Bongsoo Lee [Chung-Ang Univ.]	
6-P-10	Assessment of Cosmic-Ray Contribution on Aerial Survey Using Large Volume NaI (TI) Detector	382
	Jun Ha Kim, Jong In Byun [KINS]	
6-P-11	Evaluation of ^{40}K Concentration in Samples Containing High Concentration of ^{232}Th	383
	Eun Ji Park, Sung Han Lee, Young Hwan Park, Su Bin Heo, Seong Gyu Lim, Dong Cheol Jang, Seung Beom Seo [HANIL Nuclear]	

6-P-12	Machine Learning Approach to Radiation Exposure Doses From Consumer Products Containing Naturally Occurring Radioactive Materials	384
	Juyoul Kim [KINGS], Gi Young Han [KINS]	
6-P-13	Analysis on the Country-specific Graded Approaches to Exemption and Clearance of NORM	385
	Zuhee Woo [KINS]	
6-P-14	Distribution of Naturally Occurring Radioactive Materials in Groundwater of Jeju Island	386
	Sohyeon Lim, Chung-Hun Han, Hee-Jung Im [Jeju Natl. Univ.]	
6-P-15	Property Analysis of Aerosol Containing NORM in Cement Manufacturing Industry for Radiological Risk Assessment	387
	Jong Hyeok Park, Byung Min Lee, Seong Yeon Lee, Kwang Pyo Kim [Kyung Hee Univ.]	
6-P-16	Disposal of a Radioactive Concrete Waste Containing Uranium by the Crush-Sieving Method	388
	Jeongbo Yang, Jeongmyeong Kim, Jeonghwan Hong, Yeongjong Ju, Jungsuk Park [KEPCO NF]	
6-P-17	Optimization of Design of a D ₂ O-moderated ²⁵² Cf Source for Personal Dosimetry Performance Test	389
	Hyeongjin Kim, Yuho Weon, Munhyung Cho [KHNP-CRI]	
6-P-18	Review of Environmental Radiation Monitoring Strategy in Nuclear Emergency	390
	Bongseok Kim [KAERI]	
6-P-19	The Results of Performance Test for Whole Body Counters in Nuclear Power Plants	391
	Si Young Kim, Seok-Ju Hwang, Cheon-Woo Kim [KHNP-CRI]	
6-P-20	A Proposal for the Development Digital Twin in the Decommissioning Nuclear Power Plant Project	392
	Sanghyun Lee, Hyung-woo Seo, Gi-lim Kim, Jihwan Yu, Hyein Kim [KHNP]	
6-P-21	An Improvement of Dimensioning Machine for Irradiated Materials in Hot Cell	393
	Yunmock Jung, Seungje Baik, Youngjun Kim, Hyojik Lee [KAERI]	
6-P-22	Preliminary Analysis of Uncertainties in the Calculated Value of Radioactivity for Airborne Discharges From NPPs in Batch Mode	394
	Seok Hee Chung, Jae Hak Cheong [Kyung Hee Univ.]	

6-P-23	Review on Evaluation Method by Direct Radiation Exposure in ODCM	395
	Seok-Ju Hwang, Deuk-man Kim, Cheon-Woo Kim [KHNP-CRI]	
6-P-24	Review of the Linkage Between Accident Management Plan and Radiation Emergency Plan	396
	Juyoul Kim [KINGS], Hyeong-Ki Shin [KINS]	
6-P-25	Measurement of Alpha Radiation Dose in Uranium-using Facility	397
	Seonho Noh, Youngwoo Kwon, Youngsang Lee [KAERI]	
6-P-26	Simulations on Routine Emission of Radioxenon in the Northern Hemisphere Using LADAS Model: Case Study in 2014	398
	Kihyun Park, Byung-Il Min, Sora Kim, Jiyeon Kim, Kyung-Suk Suh [KAERI]	
6-P-27	A Study on the Evaluation of the Moving Path and Expected Annual Exposure Dose of Workers Considering the Work Zone in Movable Radionuclide Analysis System	399
	Sangheon Lee, Minho Lee, Jongsoon Song [Chosun Univ.]	
6-P-28	Evaluation of Surface Contamination Measurement Results Based on the Recommendations of the International Organization for Standardization	400
	Chaehee Kim, Insu Chang [KAERI]	
6-P-29	Assessment of Dose Rate and Radioactivity Ratio in Some Areas of Jeju Island	401
	Byungchae Lee, Jihyun Yu, Jonghoa Kim, Jangsoo Suh [Sae-An Enertech Corp.]	
6-P-30	Analysis of Risk Assessment Methodologies in Radioactive Waste Overland Transportation Risk Assessment Codes for Transportation Accident	402
	Min Woo Kwak, Hyeok Jae Kim, Seong Yeon Lee, Kwang Pyo Kim [Kyung Hee Univ.]	
6-P-31	Discussion on Calculation Procedure of Kr-85 Concentration in Gaseous Effluent and Emission Evaluation	403
	Suhyun Choi, Cheolhyun Kang, Jongil Lee, Insu Chang [KAERI]	

7분과 방사화학 [ORAL]

7-O-01	A Review of Chemical Speciation and Thermodynamic Studies of Plutonium in KAERI	407
	Hye-Ryun Cho [KAERI]	

7-0-02	Chemical Behavior of Niobium(V) in Cementitious Conditions: Solubility and Sorption	408
	Yongheum Jo [KAERI], Nese Cevirim-Papaioannou, Krassimir Garbev, Markus Fuss, Oliver Dieste Blanco, Marcus Altmaier, Xavier Gaona [Karlsruhe Institute of Technology], Karsten Franke [Helmholtz-Zentrum Dresden-Rossendorf], Benny de Blochouse [ONDRAF/NIRAS]	
7-0-03	Application of Machine Learning to Predict the Chemical Reactions at Solid-Water Interfaces	409
	Jun-Yeop Lee, Do-Hyun Kim [Pusan Natl. Univ.]	
7-0-04	Experimental Analysis of HTO Adsorption on Activated Carbon in Tritium-Contaminated Water	410
	Jei-Won Yeon, Ji Eun Yang [KAERI]	
7-0-05	Radiochemical Separation of Nb Isotopes in Radioactive Waste Samples From the Nuclear Power Plant Using Anion Exchange Resin	411
	Myung Ho Lee, Byoung Chul Song, Kyung Sik Kim, Young Jae Maeng, Choon Sung Yoo, Byoung Chul Kim [KRIST]	
7-0-06	Modified Separation Method for ^{99}Tc , ^{94}Nb , ^{90}Sr , ^{55}Fe , $^{59/63}\text{Ni}$, ^{60}Co , ^{137}Cs	412
	Heung Nae Lee, Kwy Hwan Cho, Kyu Kwang Kang, Hyeong Min Park, Jun Hwi Bang, Yun Han Lee [Korea Standards Analysis Co. Ltd.]	
7-0-07	Analysis of Light Rare-Earth Elements Using Gas-Pressurized Extraction Chromatography	413
	Jihye Kim, Namuk Kim, Sang-Eun Bae [KAERI]	

7분과 방사화학 [POSTER]

7-P-01	Validation Result of Combustion Method for Simultaneous Analysis of ^3H and ^{14}C	417
	Kyungwon Suh, Kwang Eun Lee, Jung Bo Yoo, Tae-Hong Park [KAERI]	
7-P-02	Synthesis and Characterization of Ca-Silicate Bearing Uranium Minerals ...	418
	Wansik Cha, Junghwan Park, Hye Ran Noh, Euo Chang Jung, Sang Ho Lim, Hye-Ryun Cho, Hee-Kyung Kim [KAERI]	
7-P-03	Influence of Effective Thermal Conductivity on the Radiolytic and Chemical Environment Within the CRUD Deposits	419
	Seungjin Seo, Sungyeol Choi [Seoul Natl. Univ.], HwaJeong Han, Byunggi Park [Soonchunhyang Univ.]	

7-P-04	Thermal NaCl–MgCl ₂ Salt Purification for Reducing Reactor Vessel Corrosion	420
	Wonseok Lee, Jihun Kim, Hyeongbin Kim, Sungyeol Choi [Seoul Natl. Univ.], Wonseok Yang [KAIST]	
7-P-05	Magnesium Chloride Dehydration Methods Using Thermal and Electrochemical Treatments	421
	Jihun Kim, Hyeongbin Kim, Wonseok Lee, Taeho Jang, Sungyeol Choi [Seoul Natl. Univ.], Wonseok Yang [KAIST]	
7-P-06	Corrosion Failure Analysis of the Welded Part in the Multi-purpose Molten Salt Experiments	422
	Hwa Jeong Han, Hyunjin Boo, Jonghyeon Kim, Byung Gi Park [Soonchunyang Univ.]	
7-P-07	Study on Ion Exchange Resin for Separation/Purification of Radioactive Strontium	423
	Yeong Su Ha [Osong Medical Innovation Foundation]	
7-P-08	Corrosion Behavior Analysis of Ni–Cr Alloy in Molten NaCl–MgCl ₂ With Various Metallic Chloride at 973 K	424
	Younghwan Jeon, Jungho Hur, Jaeyeong Park [UNIST]	
7-P-09	Colorimetric Analysis on the Solubility of Nd(OH) ₃ (s) Under Neutral to Weakly Acidic Conditions	425
	Ranyong Choi, Jun–Yeop Lee [Pusan Natl. Univ.]	
7-P-10	Analysis of Iodine in Various Cement Waste Particle Size	426
	Gi Yong Kim, Jung–Weon Choi, Jung Bo Yoo, Tae–Hong Park [KAERI]	
7-P-11	Assessment of Multi–Elements Analysis Ability Using ICP–OES	427
	Jihyun Kim, Kyungwon Suh, Kwang–Soon Choi, Jung Bo Yoo, Jung–Weon Choi [KAERI]	
7-P-12	Measurement of Recovery for Validation of Pretreatment Method in Solidified Cement Ion–Exchange Resin Samples	428
	Kwang Soon Choi, Gi Yong Kim, Jung–Weon Choi, Jung Bo Yoo, Tae–Hong Park [KAERI]	
7-P-13	A Development of the Quantitative Analysis Method for Np in Solutions by Various Radiometric Methods and Absorption Spectroscopy	429
	Junhyuck Kim, Hye–Ryun Cho, Jae–Il Park [KAERI]	
7-P-14	Determination of Citric Acid in Radioactive Wastes Using Ion Chromatography	430
	Hyejin Cho, Jae Il Park [KAERI]	

7-P-15	Applications of Zone-melting to NaCl-MgCl ₂ Eutectic Salt	431
	Chan-Yong Jung, Sang-Eun Bae [KAERI]	
7-P-16	Optimization of Synthetic Conditions for Titanium-containing Anderson-type Polyoxometalate	432
	Hye Ran Noh, Jong-Yun Kim, Jeongmook Lee, Sang Ho Lim [UST], Tae-Hyeong Kim, Jung Hwan Park [KAERI]	
7-P-17	Effect of Colloids on Safety Assessment of High-Level Waste Repository ..	433
	Junghwan Park, Sang Ho Lim [KAERI]	
7-P-18	High-temperature Viscosity Measurement System for Corrosive NaCl-MgCl ₂ Molten Salt Using Brookfield Rotational Viscometer	434
	Eunkyeom Lee, Chanyong Jung, Tae-Hyeong Kim, Sang Ho Lim, Jong-Yun Kim [KAERI]	
7-P-19	Synthesis and Characterization of New Hydrogen-Bonded Organic Framework With Subsequent Application for Iodine Capture	435
	Viktoriya Kim, Jeongmook Lee, Jong-Yun Kim, Sang Ho Lim [UST], Tae-Hyeong Kim [KEARI]	

1분과

핵주기정책·규제 및 비확산(Oral)

A Study on North Korea's Denuclearization Policy Suggestion Through Ukraine-Russia War Analysis

Sungmin Yang and Gyunyoung Heo*

Kyung Hee University

*onlysumu@khu.ac.kr

Since the 1992 'Joint Declaration of South and North Korea on the Denuclearization of the Korean Peninsula' was agreed, various negotiations and policies have been conducted. There were policies such as CVID, Strategic Patience, Top-Down Approach, Calibrated Practical Approach, Audacious Initiative and the Geneva Agreement, and 9.19 Joint statement by the six-party talks were signed to denuclearize Korean Peninsula. However, starting with the first nuclear test in 2006, North Korea conducted six nuclear tests to develop atomic bombs, boosted fission bombs, ICBMs, and SLBMs to enhance its weapon capabilities. In addition, nuclear security crisis on the Korean Peninsula has been rising day by day as signs of restarting the North Korea's Yongbyon 5MWe Graphite-moderated reactor were observed and the possibility of a seventh nuclear test have been increased. Since North Korea's nuclear issue has a lot of influence on international security, especially on the Northeast Asian countries, a realistic denuclearization policy that reflects North Korea's current domestic situation along with the international situation is needed.

It's been six months since Russia invaded Ukraine on February 24, 2022. The war between Russia, which has nuclear weapons, and Ukraine, which gave up its possession of nuclear weapons due to the Cooperative Threat Reduction (CTR) program known as the Nunn-Lugar program, is expected to have a significant impact on North Korea, which is considering denuclearization due to UN sanctions on North Korea. Therefore, in this study, based on the war patterns of Russia-Ukraine war, perspective on how it could affect North Korea's denuclearization is analyzed. Also, significance and limitations of the previous nuclear negotiations, the North Korea's political regime, the 'five-year strategy for North Korea's economic development' and the 'five-year plan for North Korea's economic development' were analyzed to suggest practical DPRK's denuclearization policy.

Keywords: Nuclear Nonproliferation, Russia-Ukraine War, DPRK Denuclearization Policy

Development of Energy Mix Simulation Model for 2050 Net-Zero Carbon

Pilhyeon Ju, Sungyeol Choi, and Jongho Lee*

Seoul National University (SNU)

*jhlee6108@snu.ac.kr

The crisis of climate change aroused international needs to reduce the greenhouse gas emission in energy sector. Government of South Korea formulated an agenda of carbon neutrality through announcing 2050 Net-Zero Carbon Scenario A and B in October 2021. As the power supply from renewable energy increases, it becomes a core element to take into account the daily intermittency of renewable energy in analyzing the upcoming energy plans. However, the existing yearly Load Duration Curve is insufficient for applying day and night power change in daily scale into energy mix analysis, since it derives the energy mix for whole year on the basis of classifying annual base load and peak load. Therefore, a new energy mix simulation model based on the daily power load and supply simulation is needed for the future energy analysis.

In this study we developed a new model which simulates the average power supply and demand daily (over a 24 hour period) for each season. The model calculates the excess and shortage power during day and night by integrating each energy's daily power pattern. The 2050 Net-Zero Carbon Scenario A was used for the model verification, during which the same amounts of power production from each energy source were applied: nuclear, renewable, carbon-free gas turbine, fuel cell and by-product gas. Total power demand pattern and renewable energy production pattern were drawn from the data of 2017 power production, and Pumped-storage Hydroelectricity and Energy Storage System were used as day-to-night conversion. Detailed assumptions for each energy were based on the Basis of Calculation for Net-Zero Carbon Scenario from Government. The model was verified with three cases which were divided depending on the method of hydrogen production and whether the Curtailment and Conversion Loss (CCL) of renewable energy were considered or not. Case 1 assumed production of hydrogen occurred for 24 hours while not considering CCL, had 0% relative error in comparison of total annual power production, and case 2, considering CCL, had a 1.741% relative error. Case 3 assumed production of hydrogen occurred only during daytime with excess power and CCL consideration, yielded 0.493% relative error in total amount of hydrogen production, confirming that the model sufficiently describes the Government's Scenario A with the input of total power production. This model is expected to be used for analyzing further energy mix with different ratios of each energy source, with special focus on nuclear and renewable energy sources.

Acknowledgements

This research was financially supported by Uljin County, Gyeongsangbuk-do (20210927829-00).

Keywords: Net-Zero Carbon, Energy Mix, Power Simulation, Load Duration Curve, Nuclear Energy

Development of Sample Size Calculation Code for Review of IAEA Inspection

Byung Hee Won*, Hyun-Jo Kim, Dae-Yong Song, Bong Young Kim, and Ju-Ang Jung

Korea Atomic Energy Research Institute (KAERI)

*wonbh@kaeri.re.kr

The measurement activities to evaluate material balance of nuclear material are usually performed by operator. It is because that the IAEA does not have enough manpower to carry out nuclear measurement accountancy of all nuclear materials in the world. Therefore, the IAEA should consider scenarios which facility operator tries to divert nuclear material for misuse by distorting measurement record. It is required to verify the operator's measurement data whether it is normal or not.

IAEA measures inventory items using their own equipment which is independent of facility operator equipment for verification. Since all inventory lists cannot be verified due to limited resources, the number of items to be verified is determined through statistical method which is called as sample size calculation. They measure for the selected items using their own equipment and compares with operator's record. The IAEA determines sample size by comprehensively considering targeted diverted nuclear material amount and targeted non-detection probability and performance of measurement equipment. In general, the targeted diverted nuclear material amount is considered significant quantity (plutonium: 8 kg, uranium-235: 75 kg). If the targeted non-detection probability or the performance of the verification equipment is low, the sample size increases, and on the contrary, in the case of high non-detection probability or good performance of verification equipment, even a small sample size is satisfied. It cannot be determined from a single sample size calculation because there are so many sample size combinations for each verification equipment and there are many diversion scenarios to be considered. So, IAEA estimates initial sample size based on statistical method to reduce calculation load. And then they calculate non-detection probability for a combination of initial sample size. Through the iteration calculation, the sample size that satisfies the closest to the target value is derived.

The sample size calculation code has been developed to review IAEA's calculation method. The main difference is that IAEA calculates sample size based on approximate equation, while in this study, sample size is calculated by exact equation. The benchmarking study was performed on reference materials. The data obtained by the code show similar results to the reference materials within an acceptable range. The calculation method developed in this study will be applied to support IAEA and domestic inspection activities in uranium fuel fabrication facility.

Acknowledgements

This work was supported by the Korea Foundation of Nuclear Safety (KoFONS) grant funded by the Korean government's Nuclear Safety and Security Commission (NSSC) (No. 2101071).

Keywords: Sample Size Calculation, IAEA Inspection, Physical Inventory List, Detection Probability

Development and Validation of Prototype Real-Time Verification Sealing System

Jihyun Ahn and Hee Seo*

JeonBuk National University (JBNU)

*hseo@jbnu.ac.kr

To ensure the peaceful use of nuclear energy, nuclear safeguards are applied in member states of the International Atomic Energy Agency (IAEA) under the Non-Proliferation Treaty. The two major considerations in implementing nuclear safeguards are effectiveness and efficiency. In terms of efficiency, the IAEA has a great interest in using containment and surveillance (C/S) technology to maintain continuity of knowledge. A representative means of C/S technology is a sealing system to detect tampering. The existing sealing systems used by the IAEA are of limited functionality in real-time verification purposes. To address this limitation, the present study develops a real-time verification sealing system. First, we analyzed the design requirements of a sealing system proposed by various institutions including the IAEA, the U.S. Nuclear Regulatory Commission, and a number of national laboratories and companies. Then, we identified the appropriate design requirements of this system for real-time verification. Finally, the prototype system was developed and tested based on the identified design requirements. The validation tests of the prototype system were performed for anticipated environmental conditions, radiation resistance, and safeguards functionality. Additionally, we are developing user-friendly verification software. The software validation is planned to perform for functionality, performance efficiency, and security. The next step is to develop a commercialized real-time verification sealing system based on the results of validation tests. Using this commercialized system, we plan to evaluate the performance in various actual use cases. Such a system is expected to significantly enhance the efficiency of nuclear safeguards.

Acknowledgements

This study was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (2004024-0120-CG100).

Keywords: Nuclear Safeguards, Sealing System, Verification, Real-Time

Study on the Applicability of Event Tree and Fault Tree Analyses to the Systematic Generation of Diversion Paths

Seong-Kyu Ahn*, Bong Young Kim, Dea-Yong Song, and Ho-Dong Kim

Korea Atomic Energy Research Institute (KAERI)

*skahn76@kaeri.re.kr

Even though it is emphasized to apply safeguards-by-design (SBD) concept in the early phase of the design of a new nuclear facilities, there is no clear guideline or tools for the practical SBD implementation. Generally known approach is trying to review whether there is any conflicts or shortcomings on a conceptual safeguards components in a design information. This study tries to build a systematic tools which can be easily applied to safeguards analysis.

In evaluating the safeguards system or performance in a facility, it is essential to analyze the diversion path for nuclear materials. Diversion paths, however, can be either extremely simplified or complicated depending on the level of knowledge and purpose of specific person who do analyze in the field. In the context, this study discusses the applicability of an event tree and fault tree method to generating diversion paths systematically. The essential components constituting the diversion path were reviewed and the logical flow for systematically creating the diversion path was developed. The path generation algorithm based on the facility design components and logical flow as well as the initial information of the nuclear materials and material flows was test using event tree and fault tree analysis tools. The usage and limitation of the applicability of this two logic methods are discussed and idea to incorporate the logic algorithm into the practical program tools is suggested. The results will be used to develop a program module which can systematically generate diversion paths using the event tree and fault tree method.

Acknowledgements

This work was supported by the Korea Foundation of Nuclear Safety (KoFONS) grant funded by the Korean government's Nuclear Safety and Security Commission (NSSC) (No. 2106049-0222-CG100).

Keywords: Safeguards, Event Tree, Fault Tree, Diversion Path, Algorithm

An Analysis on Security Activities of the NuScale Standard Design Review

Kookheui Kwon and Seongyoun Jo*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jerry@kinac.re.kr

Recently, about 70 Small Modular Reactors (SMRs) are being developed around the world due to various advantages such as modularization, flexibility, and miniaturization. An innovative SMR (i-SMR) is being developed in South Korea as well, and the domestic nuclear utility is planning to apply for the Standard Design Approval in 2026 after completing the basic design and standard design. Accordingly, the regulatory body is conducting research on the regulatory system for reviewing the i-SMR standard designs by referring to the IAEA and the U.S. NRC cases.

A SMR is expected to many changes not only in terms of cyber security due to new digital technology, remote monitoring, and automatic operation, but also in terms of physical security according to security systems, security areas, and vital equipment. Accordingly, related technical documents issued by the IAEA require nuclear utilities to consider regulatory requirements of security from the design phase by integrating security regulations into SMR licensing. The U.S. NRC has also identified 17 issues affecting SMR design since 2010 (SECY-10-0034), and among them, ‘Consideration of SMR security requirements’ was included as a major issue. Accordingly, the NuScale applicant conducted security assessment and design in consideration of the Design Base Threat (DBT) in the initial SMR design process through the Gap Analysis Report (2012) and the NuScale’s Security System Technical Report (TR-0416-48929), and the NRC developed the Design Specific Review Standard for NuScale (DSRS) and then reviewed the applicant’s security design process, standard design results, and testing criteria for security system (ITAAC).

This paper analyzed the case of security review activities during the NuScale standard design review, and through this, it is intended to be used in the development of domestic regulatory system for the i-SMR security review in the future.

Keywords: Small Modular Reactor, Physical Security, Cyber Security, Standard Design Application, Regulatory System

A Review on the DBT and Nuclear Damage Compensation Act

Sangcheol Hyung* and Jaeyeong Jang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*wakendragon@kinac.re.kr

It is still questionable if the Nuclear Damage Compensation Act can be applied to security accidents on nuclear facilities caused by hacking or drone attacks.

If the Act is applied, the nuclear operator shall be liable for compensation for the damage even if there is no negligence, and no other person shall be liable. If it is not, victims must prove the negligence of nuclear operators and not only nuclear operators but also suppliers must be responsible.

According to Article 5 of this Act, a nuclear operator cannot operate a reactor before signing liability insurance contract or a compensation contract or depositing in order to compensate for nuclear damage.

The liability insurance contract includes the hacking accident, but it is not applied to drone attacks since only hacking accident is included in design basis accidents. However, Article 2 of this Act defines a nuclear accident as an event that may cause nuclear damage so it can be said that the ‘event’ includes intentional attacks.

Article 3 stipulates that nuclear operators are not liable for compensation for damages caused by armed conflicts, hostile acts between countries, or civil war or rebellion. Therefore, if nuclear power plant is attacked by missiles from North Korea, the nuclear operator is not liable for compensation. And, it can be interpreted that the nuclear operator is liable for compensation for damages caused by actions of a third party that do not fall under this category.

According to the Act on Indemnity Agreement For Nuclear Damage Compensation, nuclear damage caused during normal operation is included in the scope of compensation, but damage caused by actions of third parties is excluded. In the end, damage caused by actions of third parties is included in the scope of nuclear damage, but not included in the loss compensated by the government.

According to the Act on Physical Protection, the Nuclear Safety & Security Commission is required to establish a design basis threat that is the standard for designing and evaluating physical protection systems. Therefore, it is reasonable to include the contents of design basis threats in liability insurance, or to apply the principle of no negligence liability and focus of responsibility to protect victims, if not.

Keywords: Drone, Design Basis Accident, Design Basis Threat, Nuclear Damage

1분과

핵주기정책·규제 및 비확산(Poster)

A Study on the Analysis of Design Features of SMART to Apply Safeguards-By-Design

Joo Hyung Moon^{1,*} and Donghyuk Lim²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*moonjooh@kaeri.re.kr

The purpose of the present research is to verify the design characteristics of the SMART facility for the application of the IAEA's safeguards-by-design (SBD) concept to small modular reactor (SMR) and to establish a foundation for SBD to be faithfully implemented as early as possible from the design stage. International Atomic Energy Agency (IAEA) is planning to facilitate the verification activities of inspectors by developing a safeguards approach to the reactor as early as possible and preparing a safeguards technical report (STR) before commercial operation of SMR begins. To this end, the IAEA is developing various approaches to the application of SBD to SMR with countries such as Republic of Korea, Russian Federation, China, the United States, and Canada through the Member State Support Program (MSSP).

In order to review the unique design information of SMART facilities, the only deployable SMR in Korea, and to establish safeguards from the early design stages of SMART, it is necessary to carry out the task through cooperation with the Korea Atomic Energy Research Institute (KAERI) and Korea Institute of Nuclear Nonproliferation and Control (KINAC). IAEA agreed with the KINAC and KAERI to the direction of the project and to prepare both the Design Information Questionnaire (DIQ) and the Safeguards Technical Report (STR) for SMART facilities sequentially. The DIQ is a collection of questions to understand the characteristics of the reactor facilities that must be considered in applying safeguards. The STR is a document referenced by IAEA inspectors when verifying safeguards. Those draft versions were prepared and submitted to the IAEA. After review opinions were received, additional revision was conducted.

In 2022, the IAEA holds the consultancy meeting on SBD for SMART. The purpose of the meeting is to review the draft DIQ and STR prepared by designers and discuss the future work plan of the task with designer and the task point of contact in order to safeguards can be considered at the early stage of the design. The results will be beneficial to the efficient safeguards verification activities of IAEA inspectors in the future.

Acknowledgements

This work was supported by the National Research Foundation of KOREA (NRF) funded by the Korea government (Ministry of Science and ICT) (No. NRF-2020M2D7A1079178).

Keywords: Safeguards by Design, Safeguards Technical Report, Small Modular Reactor, SMART

Investigation of Physical Protection System Design on United States Standards and Practices

Ji-Hwan Cha*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jh_cha@kinac.re.kr

Sabotage on nuclear power plants are of great national and social significance and long-term damage, the IAEA's "Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Rev.5) provides a standard direction for physical protection of their nuclear facilities in almost all member countries, including Korea and the United States.

In the United States, Federal Law 10 CFR Part 73, Sections 73.40 to 73.57 specify requirements for physical protection of nuclear power plants, performance criteria, physical protection systems and components thereof, core information, and physical protection for key activities related to nuclear power plant operations.

Accordingly, the USNRC carefully examines whether the plant meets the physical protection objectives and criteria set out in SRP 13.6.2, whether the core area/protection area is properly set up to protect against internal and external physical attacks, sabotage threats, and what design measures and facilities are being set up for these areas.

The Department of Homeland Security (DHS), established in 2002 following the 2001 World Trade Center attacks, authorized federal, local governments, and authorities National Infrastructure Protection Plan (NIPP) to protect facilities from terrorist attacks and man-made physical attacks in 2007. NIPP clarifies the great principles and governance of the physical protection of national infrastructure in the United States presented by DHS.

There are many physical protection design guidelines and technical standards for preventing attacks from terrorists or internal and external sabotage attackers, improving the viability of mitigating the damage in case of emergency, and achieving efficient recovery from such damage.

Particularly important, small-scale damage/damage at a particular location of a major facility is extended to the entire facility, resulting in asymmetrical large-scale damage, so-called "Progressive Collapse" under initial attack loads, minimizing local damage, and protecting the building's integrity through isolation from other structural components.

Consequently, this paper deal with physical protection system design on United States standards and practices for applying to physical protection system design in Republic of Korea.

Keywords: Physical Protection System Design, Department of Homeland Security, United States Standards

Application of the Facility Safeguardability Assessment Process to the Decommissioning of Nuclear Facilities in the ROK

Seungmin Lee* and Donghyuk Lim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*seungmin@kinac.re.kr

Kori unit 1, Korea's first light-water nuclear power plant, was permanently shut down in June 2017. The operator, Korea Hydro & Nuclear Power Co. (KHNP), submitted a final dismantling plan for Kori unit 1 to the Nuclear Safety and Security Commission (NSSC) in May 2021. Pursuant to this procedure, the NSSC is preparing regulations for the decommissioning stage of large nuclear facilities for the first time in the Republic of Korea. The Korea Institute of Nuclear Non-proliferation and Control (KINAC) is also considering applying regulations on safeguards. Moreover, the International Atomic Energy Agency (IAEA) developed the "International Safeguards Guidelines for Nuclear Facilities under Decommissioning" in 2021. The guidelines describe the detailed application of safeguards measures to be considered when decommissioning nuclear facilities, dismantling essential equipment, and providing relevant information to the IAEA, as well as the scope of IAEA inspections. In addition, Dr. R. Bari of the Brookhaven National Laboratory (BNL) proposed the Facility Safeguardability Assessment (FSA), a methodology that reflects facility characteristics from the design stage to ensure that designers, national regulators, and the IAEA communicate smoothly regarding safeguards measures. The FSA process derives expected problems with safeguards measures considering new nuclear facilities by analyzing the gap of safeguards measures applied to existing similar nuclear facilities. This study uses the existing FSA methodology to predict problems related to safeguards measures when decommissioning nuclear facilities and to analyze deviations from safeguards measure requirements according to IAEA guidelines. To this end, the reference facility is set as an operating pressurized light water reactor; the issues with the safeguards measures are summarized using the FSA Process; and a draft safeguards concept for nuclear facilities under decommissioning is designed. Furthermore, validity is confirmed through a simple analysis of the diversion path, and implications and lessons are derived. Through this, it is possible to anticipate new safeguards measures to be applied when decommissioning nuclear facilities in the Republic of Korea and review problems and considerations in advance.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106018).

Keywords: Decommissioning of Nuclear Facilities, Facility Safeguardability Assessment (FSA), Safeguards Measures, Safeguards by Design (SBD)

Analysis of Cyber Security Regulatory Frameworks for EP CDAs in the U.S.

In-hyo Lee*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*lih9103@kinac.re.kr

KINAC has regulated cyber security of nuclear facilities based on 「Act on Physical Protection and Radiological Emergency」 and KINAC/RS-015 “Security for Computer and Information System of Nuclear Facilities”, a regulatory guide. By that law and regulatory guide, nuclear licensees shall protect digital assets so-called CDAs, which are conducting safety, security, and emergency preparedness functions from cyber-attack. First of all, to protect CDAs from cyber-attack, licensees should identify CDAs from their assets according to the RS-015. The identification methods are provided in another regulatory guide, RS-019. To research the best practice, a reference case is selected as a U.S. case.

In this study, a comparison analysis was conducted especially focused on EP CDAs identification methodology between R.O.K. and U.S., because the regulation basis is relatively insufficient in R.O.K., and improvement plans for the cyber security regulations in R.O.K were proposed. From the analysis, it was identified that detailed methods to identify EP function are provided in NEI 10-14 “Identifying Systems and Assets Subject to the Cyber Security Rule” published by Nuclear Energy Institute (NEI), an institute of nuclear power reactor licensees. Also identified that the definition of EP function is provided clearly in NEI 10-04 based on related regulation, 10 CFR 50.47 “Emergency Plans”.

In that regulation, licensees shall follow and maintain the effectiveness of an emergency plan that meets the sixteen planning standards of 10 CFR 50.47(b). So, these sixteen planning standards correspond to the emergency preparedness functions. In NEI 10-04, scoping considerations for emergency preparedness function are provided referring to sixteen planning standards. Moreover, in that scoping considerations, planning standards, planning standard functions and 10 CFR 73.54 “Protection of digital computer and communication systems and networks” scoping guidance are provided, so, licensees identify EP CDA in their assets conveniently.

In case of R.O.K., because these sixteen planning standards are not established, there is an ambiguity in identifying EP CDAs. The only related provision is “Detailed Standards for Establishment of Emergency Plan”. To resolve the ambiguity, it is needed to analyze sixteen planning standards in 10 CFR 50.47(b) and “Detailed Standards for Establishment of Emergency Plan”. Then, should be developed ‘scoping considerations for emergency preparedness function’ based on the analysis as provided in NEI 10-04.

Keywords: Cyber Security, Emergency Preparedness, Critical System, Emergency Plan

A Study on the Changes of Safeguards Techniques and Equipment

Inchul Kim*, Hyunjo Kim, Byungdoo Lee, Sungho Lee, Jooang Jung, Jinha Choi, Munyung Ryu,
and Seongmi Han

Korea Atomic Energy Research Institute (KAERI)

*ickim@kaeri.re.kr

IAEA has the right and obligation to verify the states' commitments for safeguards under the comprehensive safeguards agreement and additional protocols. There are IAEA inspections such as PIV, DIV, RII, SNRI under these agreements. As part of the implementation of this mission, the IAEA inspectors perform the verification for the state's accounting reports related the nuclear materials such as ICR, PIL, MBR. To do well this verification, the inspectors often use non-destructive analysis, which aims to measure attributes of the items during the inspections. This kind of an activity aims to detect the missing nuclear items or wrong things in the facility using nuclear materials. In general, NDA techniques use the neutron counting and gamma ray spectrometry. Besides, IAEA also performs several verification measures as follows.

- C/S (Containment and Surveillance techniques) is to maintain the continuity of the knowledge by giving assurance that its containment remains unimpaired.
- Unattended and remote monitoring is to transmit the data from onsite of the facility through the on-line system.
- E/S (Environmental Sampling) is to detect the minute traces of nuclear materials by smearing some points in the on-site of the facility.

Nowadays, the above mentioned techniques are important ways to increase the effectiveness of the safeguards approaches reducing IAEA actual costs. To strengthen the effectiveness and improve the efficiency of safeguards approaches, IAEA always develops and adopts the techniques and equipment for safeguards. Especially, IAEA seems to be concerned with the improvement and development of the non-destructive techniques and equipment in the fields of nuclear fuel cycle. IAEA develops the new techniques and equipment through the help of MSSP (Member States Support Programs). The IAEA defines the needs of safeguards and coordinates the support programs. After the IAEA tests and evaluates the techniques/equipment developed, IAEA decides whether to use the developed techniques and equipment during the inspection by the procedure of the IAEA quality assurance. This paper aims at studying the current changes of the IAEA equipment such as DCVD, NGSS and HCES.

Keywords: Safeguards Equipment, Inspection, Accounting Reports

A Study on the Establishment of Integrated Database for Safeguards and Nuclear Export Control

Sungho Yoon*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*shyoon@kinac.re.kr

Korea Institute of Nuclear Non-proliferation and Control (KINAC) Safeguards division and Export control division operate regulation management system each other according to their work scope and characteristics. Korea Safeguards Information System (KSIS) of Safeguards division handles information for nuclear material accounting and control. Especially, accounting and declaration reports submitted to International Atomic Energy Agency (IAEA) are important information in this system. And Nuclear Import and Export Control System (NEPS) of Export control division deals with import and export information of nuclear materials and nuclear weapon trigger list items.

Establishing and operating the integrated database as sharing information between KSIS and NEPS derive merits as follows. First, the full cycle of nuclear material transfer records can be managed by collecting information on the nuclear materials from import to export or disposal. In addition, regulatory body can verify inconsistency between transfer records and account records in date, location, element, mass etc. Especially, small quantity nuclear materials are major loop hole in nuclear material accountancy system. The accumulated material transfer data will give an evidence to catch loss nuclear material. Second, sharing the information on nuclear fuel cycle related research and development activities in both divisions can utilize the information to outreach on facility subject to nuclear technology transfer for Nuclear Suppliers Group (NSG) and additional protocol declaration for Safeguards Agreement with IAEA. Third, regulatory body is easily able to manage entire import and IAEA report procedure for items subject to the Nuclear Cooperation Agreement (NCA). In present, KINAC regulation on NCA is divided to Export control and Safeguards. Export control division conducts classification imported items subject to NCA and acquires prior consent or notifies to other country. And Safeguards division report inventory list for each NCA country to the ROK government once a year. Imported NCA inventory list will be generated automatically by merging database. Then, it can be easily verified without any additional process by both divisions.

Keywords: Safeguards, Export Control, Database, Nuclear Material

A Study on Identification of Vital Areas in Spent Fuel Storage Facilities to Protect Against Sabotage

Minho Kang* and Sundo Choi

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*mhkang@kinac.re.kr

The IAEA states that in the event of sabotage, nuclear material and equipment in quantities that can cause high radiological consequences (HRC), as well as the minimum systems and devices necessary to prevent HRC, must be located within one or more vital areas. Accordingly, in Article 2 of the ACT ON PHYSICAL PROTECTION AND RADIOLOGICAL EMERGENCY, the definition of the vital area is specified, and a nuclear facility operator submits a draft to the Nuclear Safety and Security Commission to establish vital areas and must obtain approval from Nuclear Safety and Security Commission. Since the spent fuel pool and new fuel storage area are areas where nuclear material is used and stored, they can be candidates for vital areas as direct targets of sabotage. The spent fuel pool is a wet spent fuel storage facility currently operated by most power plants in Korea to cool and store spent nuclear fuel. Considering the HRC against sabotage, it is necessary to review whether spent fuel pool needs to establish a vital area. In addition, depending on the status of plant operation during the spent fuel management cycle, the operation status of safety systems to mitigate accidents and power system change, so vital areas in fuel handling building (including spent fuel pool) also need to be adjusted flexibly.

This study compares the results of the review on whether the essential consideration factors are reflected in the identification of essential safety systems and devices to minimize HRC caused by sabotage in the spent fuel storage system with the procedure for identifying the vital area in nuclear power plants. It was reviewed from the following viewpoints: Necessity to identify necessary devices to minimize the radiation effects against sabotage on the spent fuel pool, Review of necessary elements when identifying vital areas to minimize the radiation effects of spent fuel pool against sabotage, Necessity to adjust vital areas according to the spent fuel management cycle. The main assumptions used in the analysis of the vital area of the power plant need to be equally reflected when identifying vital areas in spent fuel pool. And, the results of this study are for the purpose of minimizing the radiological consequences against sabotage on the spent fuel storage system including the spent fuel pool and used to establish regulatory standards in the spent fuel storage stage.

Keywords: Spent Fuel Pool, High Radiological Consequences, Sabotage, Physical Protection, Vital Area

A Localization Strategic Study on the Export Control Programs of Sweden, Germany, France and the United Kingdom

Jae-Kwang Kim* and Seunghyo Yang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jgwang@kinac.re.kr

In August 2021, in response to the rapidly changing trade environment, including the advancement of Information Communication Technology (ICT) and its services, the European Union (EU) implemented the Dual-Use Items Control Regulation 821/2021 to introduce an Internal Compliance Program (ICP) to the EU countries. Accordingly, the exporters should comply with the regulation to strengthen their transactions review systems.

Sweden, Germany, France, and the United Kingdom have implemented ICPs and outreach activities for dual use items. In particular, France explicitly stipulates the introduction of ICP in the law to manage and supervise it. While Sweden, Germany, and the United Kingdom strengthen the supervisory authority of regulatory agencies then companies are encouraged to autonomously introduce ICPs.

Before introducing the ICP for the trigger list items (the items) to the Republic of Korea (ROK), a comprehensive export license system for them should be firstly considered based on EU Regulations. Also the comprehensive export license might be implemented by expanding the subject for the existing license on technology export of nuclear plant into the items.

The ROK does not introduce an ICP as it does not recognize a self-classification on the items in accordance with the nuclear export control law. However, in preparation for the export to the EU countries that have intentions to introduce nuclear plants, it is necessary to analyze the export control programs of Sweden, Germany, France, and the United Kingdom.

Like the programs of Sweden, Germany and the United Kingdom, the EU regulations might be adopted to reduce the regulation burden in the ROK. With the reference of Sweden, the authority could support the Export Control Manager Certification (ECMC) system accredited by civil association then its outreach activities could be diverse and extended. Basically, the ECMC system could consist of Part I, II, III and IV and an applicant could be accredited by a civil association as the ECM after completing the courses of Part I and II. The ECMC courses might be as follow;

- 1) Part I: the Basic common course for beginner
- 2) Part II: the National export control system for the items
- 3) Part III: the International export control regulations
- 4) Part IV: Re-Certification within the certain period

In this paper, we analyzed the export control programs in Sweden, Germany, France, and the United Kingdom and suggested the ECMC system that might be applied to the ROK as above.

Keywords: Export Control, Internal Compliance Program, Self-classification, Trigger-list items, Export Control Manager Certification

A Study on the Export Control Implementation in Korea Considering the Updated EU Regulation 821/2021 for Dual-use Items Transfer Control

Jae Kwang Kim* and Seunghyo Yang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jkwang@kinac.re.kr

In 2004, in order to comply with UN Security Council Resolution 1540, the European Union (EU) came into force with Regulation 428/2009 for the export control of dual-use items, which has been working to prevent the proliferation of weapons of mass destruction (WMDs). In August 2021, it amended the EU 821/2021 to include mandating the introduction of ICPs for exporters in the member countries in order to strengthen controls over the transfer of tangible and intangible technologies. The main contents are as follows;

- 1) Mandatory introduction of Internal Compliance Program (ICP): Exporters within the EU countries should introduce a transaction review procedure through the ICP in consideration of their size and organization.
- 2) Export control on Cyber Surveillance Items to protect human rights: In order to protect human rights and comply with the obligations of international human rights law, EU countries should implement export controls on cyber-monitoring items exported from customs zones in the EU. Cyber surveillance items are specially designed to monitor, extract, collect or analyze data such as biometrics through intrusion of information and communication systems or deep packet hijacking. However, items used for purely commercial programs such as billing, marketing, quality service, user satisfaction or network security are excluded.
- 3) Expansion of the Catch-all system: EU countries should utilize the catch-all system to strengthen export controls on cyber-monitoring items, including dual-use items.
- 4) Strengthening control over the cloud: Exporters and EU countries should extend the scope of intangible technology transfer, such as electronic media, fax, and telephone, outside the EU's customs territory, and apply export control regulations such as general or comprehensive licenses to cloud transmissions outside the EU territory.
- 5) Introduction of large-scale project authorization: To reduce the administrative burden on enterprises (especially small and medium-sized enterprises) and authorities when exporters with individual or collective licenses export to one or more specific end-users for the purpose of large scale projects, provided that they ensure the implementation of an appropriate level of export controls; EU countries may introduce large-scale project license systems in the form of general authorization.

Recently, there is a possibility that the ROK would export its nuclear technologies including APR1400 to the EU member countries in the midst of the EU adoption of carbon-zero policy. In this paper, we have analyzed the EU export control regulations and suggested the future direction of nuclear export control programs in the ROK.

Keywords: Export Control, Dual-use Items Regulation, EU Regulation 821/2021

A Study on Supply Chain Control Regulation of Nuclear Facilities by SSEP Functions

Seunghoon Park*, Poe il Park, Kookheui Kwon, and Chae-Chang Lee

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*shpark@kinac.re.kr

The guidelines for cyber security regulations at domestic and foreign nuclear facilities, such as KINAC/RS-015, NRC's RG5.71 and NEI 13-10, require the establishment of security measures to maintain the integrity of critical digital assets (CDAs) and protect them as threats to the supply process. According to the requirements, cyber security requirements shall be reflected in purchase requirements from the time of introduction of CDAs, and it shall also be verified whether cyber security security measures were properly applied before introduction. Domestic licensees apply measures to control the supply chain in the nuclear safety sector to cyber security policies. The safety sector supply chain control policy has areas that functionally overlap with the requirements of cyber security regulations, so regulatory guidelines in the safety sector can be applied. However, since most of the emergency preparedness and physical protection functions introduce digital commercial products, there is a limit to applying the control of the supply chain in the safety field as it is. It is necessary to apply supply chain control operator policies, procedures, and purchase requirements for each SSEP function, or to establish cyber security integrated supply chain control requirements. In this paper, based on the licensee's current supply chain control policy, the cyber security regulation plan for supply chain control according to the SSEP (Safety-Security-Emergency Preparedness) function of CDAs is considered.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KOFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106012).

Keywords: Supply Chain Control, Critical Digital Asset, Cyber Security, Nuclear Facility, SSEP

A Study on the Structure Response of Explosion Simulation Models for the Evaluation of Explosive Threats to Nuclear Power Plants

Eojin Jeon*, Wooseub Kim, Hyeseung Kim, and Sundo Choi

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*eojin@kinac.re.kr

Nuclear power plants, which are important national facilities, require special attention against the threat of terrorism using various methods. Among the terrorist threats, as structural damage and human casualties due to explosions continue to occur, interest in the blast load is increasing. However, domestic nuclear power plants do not have sufficient design requirements for protection against the threat of explosives. To prepare for the threat of terrorism using explosives, it is necessary to evaluate the physical protection performance of nuclear power plants against blast load, and to use this to improve protection performance and establish regulatory standards.

Most of the explosion-proof designs used abroad use the empirical chart presented by UFC 3-340-02 (DoD 2008), which does not take into account the effect of near-field explosions. When explosions occur inside nuclear power plants, near-field explosions occur in most cases. In this study, it was assumed that explosives were installed in the corridor inside nuclear power plants. A spherical TNT was placed in the middle of the corridor floor to simulate near-field explosions, and the structure response according to the weight of the TNT was evaluated. The corridor was modeled with a reinforced concrete material and the LS-DYNA program was used for analysis. For the explosion model, the Arbitrary-Lagrangian-Eulerian (ALE) analysis technique applying the advantages of the Lagrangian and Eulerian methods were used. By analyzing the pressure history and the degree of deformation of the structure according to the explosion, the degree of threat caused by the explosion was analyzed.

Based on the analysis of this study, physical barriers performance database (DB) using Modeling & Simulation (M&S) will be constructed by performing sensitive analysis such as representative structure shape setting, boundary conditions, material of structures, etc. The constructed DB is expected to be used to establish regulatory standards for the physical barriers of nuclear power plants related to explosives.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Physical Protection System, Modeling & Simulation, Explosives, Blast Load

A Study on I&C System Decision Process for Cyber Security Test

Inhyo Lee*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*lih9103@kinac.re.kr

Because a cyber-attack on industrial control system (ICS) and/or critical infrastructure such as NPPs is evolving, it is necessary to develop a cyber security regulation technology corresponding to the attack technology. Nevertheless, it is almost impossible to test actual I&C systems in NPPs for evaluating cyber security against new vulnerabilities or attack vectors. So, a testing environment is needed to conduct penetration tests and evaluate the overall cyber security of NPPs. For that purpose, KINAC is developing a cyber security test bed and has plans to apply the insight from the test bed to cyber security regulation in NPPs.

Conceptually, the test bed is divided into two parts, H/W parts and S/W parts. The S/W part is mainly composed of a NPPs simulator, especially APR 1400 simulator model. Originally, the I&C systems are implemented by computer codes in the simulator model, however, parts of I&C systems in the simulator are implemented by H/W in the test bed. Then, these actual H/W-based I&C systems are wired to the simulator. Because the cyber security test only can be carried out on real I&C systems (H/W-based I&C systems), it is important to decide on the scoping of HIL in the simulator. In this decision process, the purpose of testing should be considered a priority. The decision process of scoping HIL in the simulator in line with the test purpose is discussed in this paper.

For example, if the test purpose is to evaluate the consequences of NPPs induced by cyber-attacks, safety analysis results may be utilized in the decision process. Especially, if the consequences are fuel integrity in NPPs, level 1 PSA results may be used to decide the HIL scope. In that case, most of the I&C systems are safety-grade I&C systems. The number of safety-grade I&C systems is still too much so, more analysis should be accomplished to narrow down the numbers. To derive the most mitigation measures by comparing the mitigation measures in each initiating event may be one example of narrowing down the number of I&C systems. From these processes, the scope of HIL in the simulator corresponds to the test purpose may be decided.

Keywords: Cyber Security, Test Bed, HIL, PSA

A Study on the Improvement of X-ray Security Search Education for Physical Protection Personnel

Donghyun Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*dhkim@kinac.re.kr

Domestic nuclear facilities establish a physical protection system to respond to illegal transfer of nuclear materials and sabotage to nuclear materials and nuclear facilities, and operate a security search system in order to prevent the entry of controlled items into the facility. X-ray security search is also the most widely used for such security search.

Since 2018, Korea Institute of Nuclear Nonproliferation and Control (KINAC) has developed the “X-ray security screening Web-Based Training Program (XWBT)” and has been using it in the physical protection education. The XWBT contains about 700 X-ray images of the item, and can learn X-ray images by type or package of the item. In addition, trainees can practice reading the X-ray image of the item or package, looking for controlled items, and determining whether the item could be passed or opened.

However, there is a limit to Web-Based X-ray training program alone. This is because even if the same item is contained in the same bag, the X-ray image could be varied depending on the direction, angle, and other items in the package. Therefore, in addition to XWBT, X-ray reading practice education for actual luggage should be conducted in parallel.

In addition, trainees should be familiar with various images through repetitive X-ray reading practice training so that they should be able to intuitively read X-ray images and find controlled items. Therefore, securing educational time is essential to produce skilled trainees.

Korea Aviation Security Academy (KASA), which produces professional security inspectors, has established and operated a “Security search education filed for actual luggage” where trainees can pack their own bags, read X-ray images, and practice whether there are controlled items packed. In addition, KASA provides 40-hour training for security search personnel, which focuses on improving the practical skills that security search personnel must have. This study describes the current status of “X-ray Security Search” of Physical Protection Education for security personnel and presents course improvements through the case of KASA.

Keywords: Physical Protection Education, Security Search, X-ray

A Study on the Development of the Curriculum for “Disguised Weapon Identification” and “Abnormal Behavior Detection Method”

Donghyun Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*dhkim@kinac.re.kr

Recently, extreme terrorist attacks have frequently occurred around the world and are threatening the international community. It is no longer a safe zone for terrorism in our country. Therefore, domestic nuclear facilities as the highest level of national security facilities have established a physical protection system to protect facilities and lives against terrorist attacks. In addition, security search and access control are conducted for controlled items and unauthorized person.

However, with the development of science and technology, disguised weapons or homemade explosives used in terrorism are becoming very sophisticated. Therefore, nuclear facilities need to strengthen security search of weapons or homemade explosives. Since these disguised weapons or homemade explosives are difficult to find only through security search, it is also necessary to actively identify unspecified people who possess disguised weapons or do abnormal behavior. For this reason, the “Abnormal Behavior Detection Method”, which is very effective in preemptive response to potential terrorist risks, has been introduced and operated in aviation security field.

Korea Institute of Nuclear Nonproliferation and Control (KINAC) has established a “Practice Environment for Identifying Disguised Weapons” in 2020 for trainees to recognize the dangers of controlled items and to use for physical protection education.

This Practice environment has not only the basic explanation of the controlled items of nuclear facilities, but also various actual disguised weapons were displayed. It also introduces actual terrorist incidents using homemade explosives such as attempted bombing of a cargo plane bound for Chicago and the Boston Marathon bombing. And then a model of the disguised explosives actually used is displayed and used for education.

In addition, in 2022, the “Abnormal behavior detection method” education module was developed and used for physical protection education. In this module, the outline and introduction of the “Abnormal Behavior Detection Method” and “Behavior Detection Officer (BDOs)” are explained.

In this way, the access control and security search system of nuclear facilities require the overall monitoring system, not only for dangerous goods but also for identification of persons possess and carrying them. This study describes the development of the Curriculum for “Disguised Weapon Identification” and “Abnormal Behavior Detection Method” to enhance the effectiveness of physical protection education.

Keywords: Physical Protection, Disguised Weapons, Behavior Detection, Education

A Methodology for Deriving Safety Boundaries of Nuclear Power Plants Against Unmanned Aerial Vehicle Attacks

Seokwoo Sohn*, Hyeseung Kim, and Sundo Choi

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*swohn@kinac.re.kr

UAVs (Unmanned Aerial Vehicle) are a rising threat to national facilities due to their cheap price and accessibility. Incidents such as the terrorism attack in Saudi Arabia's oil facilities and the paralysis of the airport system in England's Gatwick airport shows the need for integrating CUAS (Counter-Unmanned Aerial Systems) in important national facilities.

Recently efforts have been made to evaluate the technical performance of the CUAS. Especially SNL (Sandia National Laboratory) modified the methodology used for PPS (Physical Protection Systems) to develop a performance metrics for CUAS. The performance metrics can be used to effectively analyze the facilities capability of countering drone attacks in a probabilistic way.

In this study, we managed to derive the safety boundary of a reference nuclear power plant model based on its current CUAS and protection capabilities with a simplified methodology. Based on the outermost boundary of the model, the time table of the UAS consist of 4 variables which are the assessment time, transmission time, neutralization time and the maximum vehicle velocity. Dividing the maximum velocity to the net time derived, we estimated the minimum sensing point of the CUAS which is the minimum safety boundary of the facility to safely manage the UAV attack.

Two practice cases were evaluated with the methodology which is based on the UAV groups classified by the United States DOD (Department Of Defense) that matches the classification of the UAV in Korea. Each variable was assumed to fit the process of a realistic nuclear power plant. Using the variables, we calculated the minimum safety boundary of the facility.

With the methodology introduced in this study, regulators and stakeholders can easily evaluate the capability of the facilities CUAS for a design basis UAV attack. Also it can be used as a simple tool to analyze the facilities vulnerability for specific UAV specifications and a guideline to check the protective procedures of the facility.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Counter Unmanned Aerial System, Nuclear Facilities, Physical Protection, CUAS, Drone

A Study on the Export Control Policy of the U.S. on the Conditions of IAEA Additional Protocol for Nuclear Power Plant Export

Seung-hyo Yang*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*shyang3220@kinac.re.kr

When exporting nuclear power plants to a third country, the U.S. conditions import countries to join the International Atomic Energy Agency (IAEA) Additional Protocol. At the Korea-U.S. summit, Korea also agreed to maintain equal non-proliferation standards. This paper first analyzes how the U.S. applies the conditions for joining additional protocols to export control policies.

The U.S. Atomic Energy Act is a general law in the field of nuclear power that governs both civilian and military use of nuclear power. Article 123 stipulates matters related to “cooperation with other countries.” According to Article 123, the United States must conclude a peaceful nuclear cooperation agreement with another country that stipulates nuclear non-proliferation obligations for nuclear cooperation to a “significant” extent. Article 123 of the Nuclear Energy Act presents nine conditions for signing the Nuclear Cooperation Agreement, and matters related to safeguards are stipulated in Nos. 1 and 2, and only IAEA’s Comprehensive Safeguards Agreement (CSA) is specified as requirements under the current law.

As a result of analyzing the countries of the nuclear cooperation agreements currently signed by the United States, the United States is evaluating the AP in terms of the policy as an essential item. Among the nuclear agreements with the United States, three countries, Egypt, Brazil, and Argentina do not have AP in effect. Among them, Brazil and Argentina are recognized by the IAEA as replacing the ABACC with the AP, so only Egypt is not a member of the AP. The nuclear agreement between the U.S. and Egypt was signed in 1981 before the AP existed, and all recently signed agreements were identified as AP-effective countries.

As a result of reviewing the U.S. export control laws, the U.S. did not legislate the AP as a condition for peaceful nuclear exports. Reflecting the NSG export control guidelines, AP was legislated as an export license requirement only in exporting sensitive nuclear technology (enrichment, reprocessing). However, it is confirmed that the U.S. policy applies AP entry into force as one of the main requirements for determining whether it is harmful to nuclear exports, along with the conclusion of the Nuclear Cooperation Agreement, the application of the Comprehensive Safeguards Agreement, and military alliance. The appropriate scope of application of the Additional Protocol in Korea and its application plan will be suggested through future research.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106017).

Keywords: Export Control, Additional Protocol (AP), Comprehensive Safeguards Agreement (CSA), Nuclear Cooperation Agreement, Nuclear Power Plant (NPP)

Analysis and Prospect of the Implementation of Random Inspection in ROK

Chul Heo* and Donghyuk Lim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hc0228@kinac.re.kr

Since 2015, the IAEA has been implementing inspections by changing the paradigm from a nuclear facility level perspective to state level approach in order to enhance effectiveness/efficiency, and random inspections have become a major means of paradigm conversion. From this point of view, Safeguards Division analyzed the implementation status of random inspection conducted in ROK over the past five years by type, facility, and year, and predict the future implementation direction.

A total of 106 random inspections have been implemented in ROK over the past five years (2017-2021). When classified by random inspection type, 20 Short Notice Random Inspections (SNRI), 9 Short Notice Inspections (SNI), 34 Unannounced Inspections (UI), and 43 Random Interim Inspections (RII) were performed, of which RII accounted for the largest proportion with about 40.6%. IAEA conducts customized random inspections according to the characteristics of each nuclear facility. In the past five years, 34 cases of LWR, 20 cases of Bulk Handling Facility, 22 cases of CANDU, 30 cases of random inspections were implemented at R&D facilities, with the largest proportion of LWR. In connection with this, as a result of analyzing the current status of random inspections by year, 19 cases in 2017, 21 cases in 2018, 20 cases in 2019, 19 cases in 2020, and 27 cases in 2021 were implemented, especially in 2021, an increase of about 40% compared to the previous year.

In conclusion, the IAEA is increasing the proportion of random inspections every year in line with the updated SLA, and this stance is expected to be maintained in the future. This can be seen as proof that randomized tests are showing an increasing trend even in the COVID-19 pandemic situation that has occurred since December 2019. and, In the case of LWR, it is expected that the form of random inspection may vary depending on the direction of IAEA's SLA improvement. Through these analyses, it is expected that the direction of future safeguards policy can be established.

Keywords: Safeguards, IAEA, SLA, Random Inspection

A Study on the Measurement Uncertainty for Evaluation of Material Balance in Bulk Handling Facility

Yewon Kim and Haneol Lee*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*haneol@kinac.re.kr

The Republic of Korea is implementing safeguards for domestic nuclear facilities through cooperation with the IAEA. But it is not to evaluate the material balance for the material unaccounted for, MUF in the bulk handling facility. Although the development of a material balance evaluation program is underway, there are no related regulations. The State Regulatory Authority, SRA is performing material balance evaluation, MBE on the facility based on the design information and material balance results of the facility. However, it is not possible to directly derive measurement uncertainty for the facility's measurement equipment, which is an important variable of MBE. To solve this problem, it is trying to derive a method suitable for the domestic environment by investigating the some measurement uncertainty estimation methods and analyzing characteristics of them. In this study, the traditional measurement uncertainty estimation method, GUM method and GUM-S1 method were studied and the advantages and disadvantages were analyzed.

Due to the problems mentioned above, the uncertainty quantification technique currently being used cannot be applied to the evaluation of the domestic material balance. Therefore, we are trying to apply them to the evaluation the domestic material balance through the above three methods or a combination of them appropriately.

Through this continuing study, it is expected that it will be possible to present a plan to derive measurement uncertainty optimized for the domestic MBE environment.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106015).

Keywords: Uncertainty, Material Balance, Bulk Handling Facility, GUM, GUM-S1

Development of Evaluation Software for Security System Effectiveness

Hyeseung Kim, Sundo Choi, and Sung Soon Jang*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*ssjang@kinac.re.kr

Nuclear power plants, like other national critical infrastructures, could be under the threat of terrorism or other malicious action. Thus, a nuclear power plant has a robust security system that includes security guards, sensors, barriers, access control systems, lights, and alarm stations with security procedures. However, an effective security system is hard to design because a chain is only as strong as its weakest link, and there could be a vulnerable hole even in the robust security system. Thus, an effective security system requires the evaluation of all possible scenarios.

Evaluation software for security system effectiveness assists in systematically assessing all the possible attack scenarios. Many countries developed security effectiveness evaluation software. The first software was developed by the U.S. Sandia National Laboratories in the 1980s. Now there are several commercially available software packages with a function to simulate limited-scope combat between security guards and attacking enemies. However, academic communication is comparatively weak because it may contain sensitive information on the vulnerability of nuclear power plants.

We developed original software called Tools for Evaluating Security Systems (TESS) to identify the most vulnerable path to the designated target and model the security systems of all South Korean nuclear power plants. We also used commercial security effectiveness evaluation software, AVERT, to model the same nuclear power plants. TESS was developed to verify the results of commercial security effectiveness evaluation software for the purpose of regulatory use.

For the feasibility test, we compared the results of two software with those of force-on-force (FoF) exercises in nuclear power plants. According to the relevant Act, every nuclear power plant site should perform the FoF exercises every year. KINAC was in charge of evaluating the FoF exercise and used several of its results for the study. In the results, even in some differences in detail, the two software and FoF exercises showed qualitative similarity.

Conclusively, evaluation software is a useful tool to design and/or assess the security systems of nuclear power plants. We modeled the security systems of all South Korean nuclear power plants, and compared the developed software, a commercial software and FoF exercises. The results showed qualitative similarity. We provided the results of evaluation to nuclear operators for the better security of nuclear power plants.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Security System, Security Effectiveness, Vulnerability Assessment Software

Study on Measures to Improve the Quality of Nuclear Regulation by Regulatory Authority

Jaeyeong Jang*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jjyviva@kinac.re.kr

In Korea, nuclear regulation is supervised by the Nuclear Safety and Security Commission (NSSC). The area that required expertise in regulatory activities has been entrusted to the Korea Institute of Nuclear Safety (KINS) and the Korea Institute of Nuclear Nonproliferation and Control (KINAC) affiliated to the NSSC. Such nuclear regulatory activities should be implemented fairly and objectively because they directly affect nuclear operators and the entire nuclear industry. In addition, as the recent policy trend has flowed in the direction of improving the business efficiency and quality of public institutions, a study was intended to improve the overall regulatory quality of the institution.

First, a case study was performed on the quality management system of domestic and foreign related organizations, including the Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) in the US. Through study and analysis of the system, it was attempted to derive institutional quality requirements for KINAC and to establish a plan to use the requirements as major items of Internal Quality Control (IQC) system. By analyzing the quality management systems of research institutes such as Brookhaven National Laboratory (BNL) and Electronics and Telecommunications Research Institute (ETRI) as well as regulatory authority, the characteristics of KINAC that conduct R&D necessary for regulation were reflected.

Based on this study, the quality management system leading to IQC-EQA (External Quality Assurance)-TQM (Total Quality Management) will be established continuously to improve the business quality of regulatory authority.

Keywords: Nuclear Regulation, Quality Management, Regulatory Authority, Internal Quality Control

Analysis of North Korea's Foreign Trade and Logistics Infrastructure

Han-sol Ko*, Chan-suh Lee, and Seung-hyo Yang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hsko@kinac.re.kr

From 1970 to 1990, North Korea's trade mainly carried out maritime trade through ports for countries around the world. This trend is due to UN sanctions against North Korea, and after the third nuclear test in 2016, North Korea gradually became isolated from the outside world as it expanded to include maritime sanctions in the form of comprehensive sanctions targeting an unspecified majority of North Korea. The logistics structure of North Korea is due to the logistics infrastructure left during the Japanese colonial period in the 1950s and the political structure of the country, with railroads being the main source of logistics and passenger transportation, and roads being secondary. North Korea faced rapid deterioration and lack of facilities after 1990 due to the lack of investment in facilities and improvement of the operating system despite the advanced logistics infrastructure and operating system compared to the South. In particular, the power shortage in North Korea hindered the operation of the railroad and accelerated the aging of the railroad facilities, which has continued to this day. In the 2010s, 96% of cargo transportation in North Korea's logistics infrastructure depended on railroads and roads, and ports that were developed until the 1980s have deteriorated with little investment since the economic crisis of the 1990s. Although some ports have been developed since 2010, North Korea's east and west coasts are separated and there is no function of shipping to connect ports, so it did not have a significant impact on enhancing the nation's port capacity. As North Korea's trade dependence on China has increased, North Korea has largely relied on land transport, except for some cargoes that are advantageous for marine transport, such as coal. This structure again leads to a decrease in investment in ports, and the current vicious cycle of causing problems with North Korean port facilities is repeated. In this study, North Korea's land and marine trade logistics system such as railway, road, and shipping and the trend of foreign trade due to sanctions against North Korea were analyzed. Through this analysis, it is planned to be used as a basis for developing the routes and scenarios through which major nuclear items can be transferred in the event of denuclearization of North Korea in the future.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS), using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2004026).

Keywords: North Korea, Logistics Infrastructure, Foreign Trade, UN Sanctions

A Review on the Interpretation of EDP Term for Trigger List Items

Chan-suh Lee*, Han-sol Ko, and Seung-hyo Yang

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*chans@kinac.re.kr

Nuclear Suppliers Group (NSG), one of the international multilateral export control systems, has designated Trigger List Item and Nuclear Dual-Use Item as control items. The Trigger List Item includes the “Especially Designed or Prepared (EDP)” term as control context. This term is very subjective and vague. Why this ambiguous term reflected in the control context?

The EDP term is one of the essential elements of multilateral nuclear export controls since the NPT came into effect in 1970. The EDP concept is the basis of the rules in the Trigger List of NSG Part 1. NSG control items and contexts are decided through consensus among Participate governments, and there are three main reasons for reflecting EDP term in the guideline. First, it is to expand the control network by preventing the diversion of concerned traders. Secondly, technology development is faster than the regulatory speed, and it is tough to put all the items on the list. And last, it is to control other sensitive information such as concerned traders’ shopping list. To reduce the ambiguity of EDP, the US, UK, and Canada presented their interpretation methods at LEEM, a sub-group of the NSG.

The United States and Canada have four location criteria (inside the containment building, RCS system or control, RCS system attachment, outside the containment building) and three purpose criteria (custom-made for nuclear power, whether it is an item according to the nuclear industry standard, not made to order) to determine the EDP items. The United Kingdom considers the original design intent and the purpose for which the item was originally designed. They also think about the nature of any specific or unique design features in terms of an item’s “fit,” “form,” and “function” that determine EDP item. Currently, in the case of KINAC, although it is not a controlled item in Korea, items designated by other countries are controlled as EDP items, and detailed standards for parts are prepared and controlled as EDP items.

The interpretation of TL’s EDP differs from country to country, and differences are allowed between member states to some extent. Suppose Korea, like the US, UK, and Canada, examines measures to control EDP and reflects it after benchmarking. In that case, it is expected that it will contribute to reducing the export control loopholes for advanced technologies in the future and improve nuclear transparency through strict implementation of Korea’s export controls.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS), using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2004026).

Keywords: Nuclear Suppliers Group, Trigger List Item, Especially Designed or Prepared (EDP)

Development of Semi-Automatic Satellite Imagery Analysis System to Support Monitoring of Potential Nuclear Activities

Gayeon Ha*, Jae-Jun Han, and Minsoo Kim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*gyha@kinac.re.kr

For countering nuclear proliferation, satellite imagery is being used to monitor suspicious nuclear activities in inaccessible countries or regions. Monitoring such activities involves detecting changes over time in nuclear facilities and their surroundings, and interpreting them based on prior knowledge in terms of nuclear proliferation or weaponisation. Therefore, analysts need to acquire and analyze satellite images periodically and have an understanding of nuclear fuel cycle as well as expertise in remote sensing. Meanwhile, as accessibility of satellite information has been increasing and accordingly a large amount of high-resolution satellite images is available, a lack of experts with expertise in both fields to perform satellite imagery analysis is being concerned. In this regard, the Institute of Korea Nonproliferation and Control (KINAC) has developed a prototype of semi-automatic satellite imagery analysis system that can support monitoring of potential nuclear activities to overcome the limitations of professionals and increase analysis efficiency. The system provides a satellite imagery database that can manage acquired images, and the users can load images from the database and analyze them in stages. The system includes a preprocessing module capable of resizing, correcting and matching images, a change detection module equipped with a pixel-object-based change detection algorithm for multi-temporal images, and a module that automatically generates reports with relevant information. In particular, this system continuously updates open-source information database related to potential nuclear activities and provides users with an integrated analytics platform that can support their interpretation by linking related images and textual information together. As such, the system could save time and cost in processing and interpreting satellite images by providing semi-automated analytic workflows for monitoring potential nuclear activities.

Acknowledgements

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

Keywords: Satellite Imagery, Change Detection, Open-source Information, Semi-automatic Analysis System

A Review on the Management System for Items Subject to a Nuclear Cooperation Agreement Owned by Nuclear Licensees

Hee Su Choe and Seung-hyo Yang*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*shyang3220@kinac.re.kr

Under the bilateral nuclear cooperation agreements (NCA) and its administrative arrangement (AA), Korea annually exchanges the inventory of subject items (including nuclear materials (NM), non-nuclear materials, equipment, and related information) with US, Canada and Australia. Also, the government performs export control procedures such as notification or prior consent during importing and exporting of relevant items. It makes NCA a means of realizing the nuclear non-proliferation regime. However, it raises difficulties in management because the entity that uses and treats those items are end-user, not the authorities of AA, the government agency of each country.

Accordingly, to increase the accuracy and effectiveness of item management at the national level, it is required to establish a system for the individual company that has the NCA items, considering the characteristics of each company. In this study, significant companies are classified into more than three types, and the management system of the items subject to the agreement is analyzed.

Each company's item management status has different characteristics depending on its role (position) within the entire nuclear fuel cycle, the type of facility, its possessed items, the main form of national trade, and the frequency of domestic movement. Those differences lead to diversity in the management systems currently owned by each company.

For example, from the perspective of nuclear materials, institutions requiring bulk management have systematically organized their management system and obligation code program compared to the 'item institutions' that can track batch history for all facility inventory changes. Although Domestic law imposes only the duty of origin management on NMs, fuel manufacturers or research institutes have established their standard obligation codes to manage multiple obligations.

The non-nuclear materials and equipment can be easily tracked and controlled by individual items. However, the management of NCA items is a complicated task involving various processes, from importing goods to using, storing, managing inventory change, selling to others, or fulfilling the obligations of AA when exporting. In particular, when the movement of items within a company or international trade occurs frequently, or when the end-users are diverse, the management difficulties increase. So a system that can accurately convey and track items subject to the AA is needed. In addition, since various entities are related, it is necessary to improve understanding of NCA items to increase the system's utilization and effectiveness.

The comparison result and requirement for system improvement based on the review above will be reflected in the history management system for items subject to NCA under development.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONs) using the financial resource granted by the Nuclear Safety and security Commission (NSSC) of the Republic of Korea (No.2106017).

Keywords: Nuclear Cooperation Agreement, Administrative Agreement, Management System, Nuclear Licensees

The Analysis of Major Proliferation Resistance Barriers in the Dry-processing of Spent Fuels for an Efficient Final Disposal

Won Il Ko*, Hyo On Nam, and Yung-Jun Cho

Korea Atomic Energy Research Institute (KAERI)

*nwiko@kaeri.re.kr

There are highly toxic radio-isotopes and high heat emitting isotopes in spent nuclear fuels which could be a burden in a deep geological repository. Some preliminary study in order to see if there are some advantages in terms of waste burden, in case that the spent fuel is appropriately processed and then disposed of in a final repository, has been carried out at KAERI. This study is focused on the proliferation resistance for various processing alternatives for them. The evaluation criteria and their indicators for proliferation resistance analysis are selected and then evaluated quantitatively or quantitatively for the alternatives.

The processing alternatives are grouped into three categories according to the level of decrease of burden for final disposal and named them as Level I, Level II and Level III technology alternatives. Level I alternative is to maximize the long-term safety in the final repository from the removal of I-129, semi-volatile radioisotope, which is the greatest impact on the long-term safety of the repository. Level II alternative is to remove the strontium-90, high heat emitter, in addition to the removal in Level I. The Level III is to additionally remove uranium from main stream of the level II to reduce the volume of the high level wastes to be disposed.

The intrinsic radiation and chemical barriers against the nuclear proliferation are selected and analysed for the alternatives. It is resulted from the proliferation resistance analysis that all three options showed excellent resistance to nuclear proliferation for the two barriers. However, Level III technology including electrochemical refining process is relatively a little weaker than others. Overall, it could be an effective means to reduce the burden of disposal if the spent fuels are appropriately conditioned for final disposal. Further detailed studies are, however, needed to finalize its feasibility.

Keywords: Dry-processing, Spent Fuel, Proliferation Resistance, Non-proliferation, Radiation Barrier, Chemical Barrier

A Study on the Establishment and Utilization of National Nuclear Technology Information Collection and Analysis System

Hyun-doo Kim and Seung-hyo Yang*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*shyang3220@kinac.re.kr

By Foreign Trade Act and the relevant laws, any organization which intends to export strategic items shall be classified whether the products, S/W, or technology are strategic items and obtain an export license from the Nuclear Safety and Security Commission (NSSC) in case those are nuclear-related strategic items.

The NSSC and KINAC conduct outreach activities to enhance the recognition and implementation of nuclear export and import control. However, ensuring that all target organizations identify and comply with relevant legislation and laws is difficult. Therefore, the government and regulators should continuously make efforts to reduce blind spots for nuclear export and import control. KINAC established the National Nuclear Technology Information Collection and Analysis System (NNTICAS) to find nuclear-related companies through online website information of various companies and to list organizations that handle nuclear-related strategic items.

Based on the NICE corporate information website, NNTICAS finds related companies using dynamic collection methods through interactions such as search word input, search button input, check box click, and collected text from each company website using static collection methods through URL address access and HTML source code extraction. After that, the text contained on the company website is analyzed to check whether the predefined normalized word is included, and if the normalized word appears at a certain number or higher, it is judged as a potential target company dealing with nuclear-related strategic items. Information on the potential target company is compared with the export controlled item list and finally classified as a target company dealing with nuclear-related strategic items.

According to the comparison a manually analyzed result of potential target companies and finally selected target companies NNTICAS, it is confirmed that predefined normalized words need to be more detailed, and a controlled item list such as axes (0B001.b.2) needs to be additionally processed. In addition, if the collection is repeated without completing the previous classification of the collected corporate information, it is also possible to collect companies in cosmetics, semiconductors, and displays.

Although it was confirmed that some supplementation is needed to use this system to select target companies for the outreach activity, it is also confirmed that target companies for outreach activity can be expanded through this system and is expected to increase the implementation rate and reduce the blind spot for nuclear export and import control.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106017).

Keywords: Nuclear Export, Nuclear Import, Outreach, Artificial Intelligence

The Requirements Analysis and Concept Design for Development of Obligated Items Management System

Mun-Young Ryu*, In-Chul Kim, and Seong-mi Han

Korea Atomic Energy Research Institute (KAERI)

*myryu@kaeri.re.kr

The bilateral nuclear cooperation agreement provides the basis for technical and economic exchanges and research and development cooperation in the nuclear industry, while also having an important role in the peaceful use of nuclear power. The government of Republic of Korea has signed the supplementary and additional arrangements with the United States, Canada, and Australia to specify detailed procedures necessary for the implementation of the agreement.

Currently, items under the bilateral nuclear cooperation agreement (i.e., obligated items) are managed through official documents and e-mails, and it is difficult for the government to systemically track item inventory and identify the person in charge. Another issue is the discrepancy of information between the annual report on obligated items and the report under the import and export procedure. In order to solve these problems, the government is establishing an import and export management system for obligated items to enhance transparency and national reliability in Korea. The ultimate purpose of this project is to not only faithfully fulfill the obligations under the bilateral nuclear cooperation agreement, but also to increase the efficiency of implementation work through systematization of databases and management of obligated items.

In this paper, an improved management procedures are suggested by analyzing the required functions and problems. Also, this paper provides a conceptual design that can be applied to the management system for obligated items by identifying and analyzing practical problems in the import and export management of mandatory items.

As a result, key considerations and the conceptual design proposal for the ‘export and import management system for items subject to the nuclear bilateral cooperation agreement’ are derived. The result will be supplemented through continuous discussions with facility operators and the government to be utilized for detailed system design.

Keywords: Nuclear Bilateral Cooperation Agreement, Obligated Items Subject to the Agreement, Management System Design, Database for Obligated Items, and Export and Import Control

Development of Installation Standards and Evaluation Indicators for Passive Infrared Sensor in National Security Facilities

Hyeseung Kim* and Sundo Choi

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*khs0301@kinac.re.kr

For national security, approximately 500 facilities have been designated as national security facilities. It is divided into grades A, B, and C depending on the fields such as electrics, energy and public service. In 2016, KINAC developed and suggested the installation standards and evaluation indicators for security equipment for national security facilities and they have been utilized to evaluate and manage the facilities. In order to update them by reflecting the recent security trend of the facilities and new technology, KINAC have investigated and updated the installation standards and evaluation indicators for security equipment.

As a part of developing installation standards and evaluation indicators for national security facilities project, we conducted survey to identify the status and capabilities of facilities: how the security system is well designed, which type of security equipment is utilized. According to the results, the passive infrared (PIR) sensor is commonly used for interior intrusion detection (not included in 2016 project). Therefore, we investigated the passive infrared sensor and provided its installation standards and evaluation indicators. PIR sensor detects the energy generated by the objects which is different as their temperatures. PIR sensors do not emit any energy like active infrared sensor, and do not measure the energy. Rather, PIRs measure variation in thermal radiation. PIR sensor detects thermal energy by sensing the change between a heat source and the background temperature.

In this paper, the characteristics and features of PIR sensor were discussed. Considering them, the standards and indicators to install/evaluated the PIR sensors were also proposed. The results would support national security facility to design and evaluate their security system.

Keywords: Security, National Security Facility, Passive Infrared Sensor, Installation Standard, Evaluation Indicator

A Study on the Response for Drone Terrorism in Nuclear Facilities by Analyzing Statistical Data of Drone Incidents

Sundo Choi*, Hyeseung Kim, and Seokwoo Sohn

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*sdchoi@kinac.re.kr

As drone technology and industry develop around the world, the use of drones are increasing in number and expanding to different fields. On the other hand, illegal flight and terrorist incidents using drones are also increasing day by day. In Korea, it is reflected in the “Design Basis Threat (DBT)”, which is the standard for designing and evaluating the physical protection system of nuclear power plants in accordance with the “Act on Physical Protection and Radiological Emergency”, that nuclear power plants continue to establish physical protection against drone threats.

A total of 141 drone attacks or incidents have occurred around the world since 2015. Cases related to the Russian-Ukraine war, in which so many cases occurred, were excluded. There were 112 cases (79%) of terrorism or suspected terrorism using a single drone. There were 4 cases of terrorism using more than 5 drones, and a total of 20 drones were used to attack an oil facility in Yemen (2019). By region, a total of 111 incidents occurred in Middle East & North Africa. By country, there were 49 cases in Iraq, 35 cases in Saudi Arabia, and 8 cases in Syria. Among major countries, three cases occurred in Korea, five in the United States, two in England, Canada, and Italy, and one in Japan and France. Since 2021, there have been 15 drone attacks. Multiple drones were used in attacks targeting military or large-scale Important National Facilities such as the Saudi oil refinery, Indian Kashmir air base, and reconnaissance of Iranian Natanz nuclear and surrounding military facilities. Also in 89% of the cases, the drones were loaded with explosives in order to cause large-scale damage.

Accordingly, nuclear power plants, which are important national facilities, need to establish a system that can detect and respond to multiple drones. Furthermore, additional protective measures are needed for areas that are expected to be severely damaged which can be established by evaluating the impact of explosives on major points in the plant. In addition there is a high possibility of terrorism by organizations aiming for national turmoil rather than individual terrorists. So it is important to identify signs of terrorism in advance and prepare through cooperation with related agencies.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Drones, Terrorism, Sabotage, Physical Protection

A Study on the Performance Test of Biometrics for Counterfeit Technology in Nuclear Facilities

Sundo Choi* and Hyeseung Kim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*sdchoi@kinac.re.kr

In order to enter a nuclear power plant, access approval is required in advance, and biometric information such as fingerprints of visitors must be registered when issuing a key card, and only those certified through biometric equipment can enter the nuclear facilities (Protected area II). Fingerprint recognizers and facial recognizers are installed and operated in domestic nuclear facilities for access control. Domestic nuclear facilities establish and implement a protection system in accordance with physical protection requirements under the “Act on Physical Protection and Radiological Emergency” and “Physical Protection Regulations” of each nuclear facility. Detailed implementation standards are specified in Regulation Standard (RS) documents established and distributed by KINAC. Biometrics are mentioned in a KINAC RS-104 (Access Control) document.

In this study, it was analyzed what points should be considered in order to prepare for performance tests and establish plans for biometric devices. In order for the results of performance evaluation of biometric devices to obtain high reliability and to be applied to nuclear facilities in the future, standardized performance evaluation targets, procedures, standards, and environments must be created.

In order to collect samples such as fingerprints for performance evaluation, the size roll of the sample shall be determined, and the appropriateness of the sample size shall be evaluated in consideration of reliability and error range. In addition, the analysis results for the characteristics (gender, age, etc.) of the sample should be presented. When collecting samples, conflicts with other laws such as personal information protection should be considered, and the reliability of the performance test result data should be analyzed and presented. Quality evaluation should also be performed on forged biometric information data such as silicon fingerprints. In addition, when establishing a performance evaluation plan, a systematic evaluation procedure should be established by referring to domestic and foreign certification and evaluation systems such as the Korea Internet & security Agency (KISA).

In order to improve the completeness of the access control system using the biometrics of nuclear facilities, it is necessary to test the performance of biometric devices and to install and operate only devices that have the ability to defend against counterfeit technology. In this study, it was analyzed what points should be considered in order to prepare for performance tests and establish plans for biometric devices.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Biometric, Performance Test, Physical Protection, Counterfeit Technology

Using Textual Analysis of News Articles to Identify State's Nuclear Activity

Hojung Do*, Dongjin Kim, and Minsoo Kim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hojungd@kinac.re.kr

Monitoring a state that intentionally hides its nuclear activity via open-source information is akin to looking through a black box. Direct information on the state's nuclear activity remains in the dark, leaving scholars to speculate how much nuclear material or warheads are being produced. Nevertheless, a state's nuclear program consists of a complex network that ranges from producing weapon-grade nuclear materials by operating its nuclear facilities to securing resources to fund these activities. These indirect activities allow a narrow window of opportunity for researchers to map a state's activity that sometimes may not be directly linked to nuclear activity per se but is significant to maintaining and operating its nuclear program. These may include malicious cyberattacks to steal or launder cryptocurrency and facilitating cooperation with fellow rogue states that do not comply with the NPT and nuclear nonproliferation regime.

The problem lies in how researchers can map this network. Much of the literature that uses text analysis uses data from either (1) formal statement, reports, and documents or (2) journal articles to extract relations between topics that is otherwise difficult to surmise. This study, however, analyzes news articles containing keywords related to a states' nuclear activity such as international sanctions, trade activities, other states' policy etc. While news articles fail to live up to the academic rigor of journal articles and unlike formal documents may sometime contain misinformation or incorrect facts, they are a valuable medium to show the day-to-day activity of a state. Although bias may exist as particular news articles may or may not be chosen for text analysis, by using articles collected from 2021 to 2022, this study argues it is enough data to show a short-term trend in nuclear activity.

Keywords: Text Analysis, Nuclear Activity, News Articles

A Study on the Effect of Frequency Scaling Factor on FBP

Jisu Kim¹, Jihun Moon¹, Heejun Chung¹, Kyung Taek Lim², and Sung Woo Kwak^{1,*}

¹Korea Institute of Nuclear Nonproliferation and Control (KINAC)

²Sejong University (SJU)

*swkwak@kinac.re.kr

For safeguarding dry storage facilities, a tomography system based on fast and thermal neutron detection was studied in Korea Institute of Nuclear Nonproliferation and Control. The study was conducted laboratory-scale experiments based on a custom built 1/10th scale model cask, He-4 gas scintillation detector array, and multiple ²⁵²Cf sources. A filtered back projection (FBP) was utilized to obtain the cask image via MATLAB. The Ram-Lak filter (ramp filter) was employed in FBP for improved the reconstructed image quality. The Ram-Lak filter is the increasing amplitude filter due to the increasing spatial frequency of the image. In spatial frequency, the frequency of brightness change in the low-frequency region is relatively low, and the frequency of brightness change in the high-frequency region is large. Thus, the high-frequency region in the neutron tomographic image is near the neutron sources and the cask, and the low-frequency region is outside of the cask and/or between the source and cask in the study.

In order to apply the ramp filter, a Fourier transform is initially performed on projection data, and image reconstruction is performed with the corrected projection data. In this case, the filter is linearly changed. Therefore, a small filter value is applied at lower spatial frequencies to reduce the projection data, and a large filter value is applied at high spatial frequencies to reduce the projection data. The filter scale is a fraction of frequency amplitude, and the filter value applied to the projection data is determined according to the filter scale. This study was conducted for discussion of the image quality due to the effect of the filter scale used for image reconstruction of a neutron tomography system.

The results show that in the experiment with one source, the source location was founded when we used the frequency scale of 0.5 and over. In the double or triple source experiment, the source locations and relative activities were found when we used a filter scale of 0.4 to 0.6. When the filter frequency scale of 0.7 to over, the relative activities are hard to know. It can be found that if the filter value is too large or too small, distortion may occur in the reconstruction results. Therefore, it seems reasonable to set a value between 0.4 and 0.6 as the scaling factor for the neutron tomography system. In the future, additional comparative studies will perform validation of the frequency scaling methods.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106016).

Keywords: Safeguards, He-4 Gas Scintillation Detector, Filtered Back-projection (FBP), Filter Scale

A Study on the Analysis of Research Outputs of KINAC (2010-2021) and Support Methodology for Performance Diffusion

Hyun Young Kim and Heejun Chung*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*hjchung2@kinac.re.kr

Since the National Research and Development Innovation Act. was enacted, evaluation and management of research outputs become more important. The research output is defined as all types of information resources produced from the scientific research activities in each research phase, such as learning, proposing, performing, and publishing. Among them, research outputs mainly from the publishing phase have been systematically managed, including articles, books, technical reports, patents, and software. KINAC, like other Korean R&D institutions, has also achieved growth in research outputs, through continuously increased investment in R&D projects. However, R&D productivity, technology transfer, and commercialization remain low level.

The importance of R&D performance diffusion has been emphasized. It's because the creation of economic value through the utilization of research results has been emerging as a key issue in R&D policy. Therefore, various policies are being nationally pursued to promote the utilization of research achievements, but the results are not being effectively utilized and disseminated. In the field of nuclear nonproliferation and security, it is also difficult to diffuse the R&D performance.

In this study, the research outputs of KINAC from 2010 to 2021 were analyzed. A number of research outputs have been made and managed, but the R&D performances have not been analyzed and identified yet. In addition, supportive methods were suggested for efficient performance management toward diffusion. For this purpose, some policies of Korea and other countries concerning performance diffusion -related policies were reviewed. The best practices of performance management and applications were also reviewed and compared to the KINAC cases.

As a result, the number of research output has dramatically increased during the last 10 years, showing an average annual total output of 84 and year-on-year increase of 18%. The biggest change was in the conference papers and journals, whereas, there were no trends by year in the case of the technical reports and patents. Of course, it was proportional to the size of the organization, the number of research projects, and its budget. Because many studies highlight the importance of institutional resources, capabilities, and processes for performance management as factors affecting efficient diffusion, the current status of our process was also identified. This study is expected to be applied to the improvement of the performance management of the institute, leading to the enhancement of the R&D performance application.

Keywords: Research Outputs, R&D Performance, Performance Management, Performance Diffusion

The Necessity of Establishing a Management System for Obligated Items Subject to Bilateral Cooperation Agreement

Seongmi Han*, In-Chul Kim, and Mun-Young Ryu

Korea Atomic Energy Research Institute (KAERI)

*Seongmihan@kaeri.re.kr

The main purpose of the Bilateral Nuclear Cooperation Agreement is to obtain the prior consent of suppliers in the case of peaceful use of items covered by the agreement, application of IAEA safeguards, reprocessing, enrichment or transfer to a third country. Reports on inventory changes and status for mutually transferred obligated items should be exchanged annually.

According to the Agreement, items subject to bilateral agreement information must be exchanged with each other prior to direct or indirect transfer of controlled items. And the importing country proceeds with prior confirmation. After that, upon receipt of the target item, shipment notification and shipment confirmation are made, and an annual report on the target item is made.

Such as the Korea Atomic Energy Research Institute (KAERI), annual reporting and management of obligated items are made centered on institutions that use a lot of nuclear materials. But there are cases of delays in the agreement work due to the implementation, and discrepancies in data are occurring in the process of checking inventory details of obligated items. In addition, it was difficult to check the inventory of items subject to the agreement and the status of Export and Import status online, making it impossible for managers to monitor all aspects of bilateral agreements.

Currently, there is generated to inconsistent in information between the annual report and the international transfer report in terms of Export and Import control. To solve these problems, KAERI is aim of promoting transparency in the international nuclear power sector and enhancing national reliability. And It is planning to establish an Export and Import management system for items subject to bilateral. In order to ensure the accuracy, it is going to enhance the efficiency of management methods such as new registration for new institutions when exporting and Importing items.

This has the ultimate purpose of improving the efficiency of the implementation of the agreement items through the systemization of the database of agreement items and the management of the implementation of the agreement based on the sincere and timely implementation of the agreement.

Keywords: Export and Import Control, Bilateral Nuclear Cooperation Agreement, Management System

Monte Carlo Simulation for Designing a CZT-based Fork Detection System

Wooseong Hong and Geehyun Kim*

Seoul National University (SNU)

*gk.rs@snu.ac.kr

The detector response was simulated to design a fork detection system for verifying the characteristics of spent fuel. The fork detection system currently used consists of two fission chamber and an ion chamber, and it is nuclear safeguard equipment that measures the gross neutrons and gross gamma rays emitted from the spent fuel assembly to identify the characteristics of the spent fuel and verify the authenticity of the operation history. In order to improve the current fork detection system, we are developing a system that applies CZT, a room temperature semiconductor detector, and a stilbene detector, which is an organic scintillator.

Depletion calculations were performed using the ORIGEN code to determine the radiological characteristics emitted from spent nuclear fuel assembly. The flux of radiation emitted from the spent nuclear fuel assembly was calculated by changing the conditions such as initial enrichment, burnup, and cooling time, which are major variables of spent fuel assembly. The calculated result is used as the source term of the particle transport code. Considering the general operating conditions of the pressurized light water reactor, the conditions were changed in the range of 3-5% for initial enrichment and 30-72 GWD/MTU for burnup, and the cooling time was given within 10 years. MCNP 6.2, a Monte Carlo simulation code, was used to simulate the detector response to radiation emitted from spent nuclear fuel assembly. According to the shape, size, and position of the CZT detector, the gamma counts incident on the detector were calculated and derived the initial design of our fork detection system.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (KOFONS2204020-0122-SB110).

Keywords: Fork Detection System, Detector Response, CZT, Monte Carlo Simulation

Analysis of Nuclear Material Distribution Behaviors Under the Abnormal Conditions of the Nuclide Management Technology

Bong Young Kim*, Young Eun Jung, Se-Hwan Park, and Seong-Kyu Ahn

Korea Atomic Energy Research Institute (KAERI)

*bykim@kaeri.re.kr

The nuclide management technology for separating high-heat generating/high-mobility/long-lived nuclides from high-level wastes based on the chemical reactions is under development. In order to secure the reliability of nuclear non-proliferation and to implement the effective safeguards, it is necessary to consider the safeguards from the conceptual design phase of the novel technologies. However, there was no experience and research on safeguards for the chemical reaction based nuclide management technology.

In order to development the available monitoring techniques for the safeguards of nuclide management technology, the possible diversion scenarios were developed and the material flows of major nuclear materials were analyzed according to the various diversion strategies for each unit process in this study. The diversion strategies in this study is limited to the diversion of nuclear materials according to the change of operational parameters (temperature, chemical reagents, pressures, etc). The nuclear material distribution behaviors under the abnormal conditions were analyzed and compared with normal conditions using the HSC Chemistry.

The results will be used to determine the proper signals and feasible techniques to monitor the abnormal operations.

Acknowledgements

This work was partly supported by a National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP) (NRF-2021M2E3A3040093).

Keywords: Safeguards, Monitoring, Nuclide Management Technology, Signal

A Study on the Application of Safeguards to Small Modular Reactor

Jinha Choi , Hyun-Jo Kim*, Byung-Doo Lee , Sung-Ho Lee , and Juang Jung

Korea Atomic Energy Research Institute (KAERI)

*keiki@kaeri.re.kr

In addition to Korea, various countries such as the United States, the United Kingdom, France, and China are designing small module-type reactors. In particular, a small modular reactor is the power of 300 MWe or less, in which the main equipment constituting the nuclear reactor is integrated into a single container. Depending on the purpose, small modular reactors are being developed to help daily life such as power, heating supply, and seawater desalination, or for power supply such as icebreakers, nuclear submarines, and spacecraft propellants.

Small modular reactors are classified according to form. It can be classified into light-water reactors/pressurized light-water reactors based on technology proven in commercial reactors, and non-light-water reactors based on fuel and coolant type such as Sodium-cooled Fast Reactor, High temperature gas-cooled reactor, Very high temperature reactor and Molten salt reactor.

SMRs, which are designed for various purposes, have the biggest difference from commercial nuclear reactors. The size of SMRs is as small as 1/5 of that of the commercial reactors. Several modules may be installed to generate the same power as commercial reactors. Because of the individually operation for each module, load follow is possible. Also, The reactor can be cooled by natural convection because the size is small enough. It is manufactured as a module, the construction period can be reduced.

Depending on the characteristics of these SMRs, application for safeguards is considered. There are many things to consider in terms of safeguards. Therefore, it is IAEA inspection or other approaches for SMRs installed and remotely operated in isolated areas, data integrity for remote monitoring equipment to prevent the diversion of nuclear materials, verification method and material accountancy and control for new fuel types and reactors.

Since SMR is more compact and technical intensive, safeguards should be considered at the design stage so that safeguards can be efficiently and effectively implemented, which is called the Safeguards by design (SBD) in the IAEA.

In this paper, according to the characteristics of SMR, we will analyze the advantages/disadvantages from the point of view of safeguards and explain what should be considered.

Keywords: Safeguards, IAEA, SMR, Safeguards by Design

Development of Process Flow Simulator of Uranium Fuel Fabrication Facility for Material Balance Evaluation Analysis

Byung Hee Won*, Hee-Sung Shin, Ho-Dong Kim, and Jinha Choi

Korea Atomic Energy Research Institute (KAERI)

*wonbh@kaeri.re.kr

Material balance evaluation is an important measure to determine whether or not nuclear material is diverted. A prototype code to evaluate material balance has been developed for uranium fuel fabrication facility. However, it is difficult to analyze the code's functionality and performance because the utilization of real facility data related to material balance evaluation is very limited. It is also restricted to deliberately implement various abnormal situations based on real facility data, such as nuclear diversion condition.

In this study, process flow simulator of uranium fuel fabrication facility has been developed to produce various process data required for material balance evaluation. The process flow simulator was developed on the basis of the Simulink-SimEvents framework of the MathWorks. This framework is suitable for batch-based process modeling like uranium fuel fabrication facility. It dynamically simulates the movement of nuclear material according to the time function and provides process data such as nuclear material amount at inputs, outputs, and inventories required for Material Unaccounted For (MUF) and MUF uncertainty calculation. The process flow simulator code provides these data to the material balance evaluation code. And then the material balance evaluation code calculates MUF and MUF uncertainty to evaluate whether or not nuclear material is diverted. The process flow simulator code can simulate the movement of nuclear material for any abnormal situation which is difficult to implement with real process data.

This code is expected to contribute to checking and improving the functionality and performance of the prototype code of material balance evaluation by simulating process data for various operation scenarios.

Acknowledgements

This work was supported by the Korea Foundation of Nuclear Safety (KoFONS) grant funded by the Korean government's Nuclear Safety and Security Commission (NSSC) (No. 2106069-0222-CG100).

Keywords: Process Flow Simulation, Material Balance Evaluation, Uranium Fuel Fabrication Facility, Material Unaccounted for, MUF Uncertainty

Evaluation of Blast Loads on a Reinforced Concrete Wall Under Internal Explosion

Wooseub Kim* and Hyeseung Kim

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*wskim@kinac.re.kr

In response to the increase in international terrorism threats and demands for terrorism prevention and response activities, the Act on Counter-Terrorism for the Protection of Citizens and Public Security was enacted in 2016, and the need for research to strengthen protection against explosive threats was raised. In the same manner, the Design Basis Threats, which become the standard for the design and evaluation of physical protection systems for nuclear facilities, have been developed and it includes explosive threats. However, the regulatory standards for physical barriers against explosive threats are still not established. Therefore, it is first required to set up a performance database of physical barriers subject to blast loading in order to prepare the regulatory standards. In this study, the pressure with the trinitrotoluene (TNT) charge weights of 0.5-2 kg as a function of time was calculated using Ansys Autodyn software by assuming that the TNT is used for malicious purposes and is attached to a reinforced concrete (RC) corridor wall. The shape of the corridor was the 3×3×6 m cuboid with a rectangular hole of 1.78×1.78×6 m. The RC walls, which make up the corridor, contained the reinforcing bars with a spacing of 0.229 m and a diameter of 0.036 m. The spherical charge of a TNT was placed 0.2 m away from a RC wall in the middle of the corridor. To measure the reflected pressure after the internal explosion with a TNT, three pressure gauges were installed on the three sides of the RC walls in the middle of the corridor, respectively. The results showed that the peak reflected pressure on a RC wall with the standoff distance of 0.2 m was about ten times higher than the opposite RC wall with the standoff distance of 1.58 m in the same condition of TNT charge weight. Thus, it was verified that blast loads are highly affected by standoff distance. It seems that preventing the explosive detonation close to a physical barrier is strategically important to maintain the integrity of the physical barrier.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS) and the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (Grant No. 2106014).

Keywords: Explosive Threats, Physical Barrier, Blast Loads, Ansys Autodyn

A Study on Request of Governmental Assurance for the Export of Trigger List Items

Su-Hyeon Kim* and Seung-hyo Yang

Korea Institution of Nuclear Nonproliferation And Control (KINAC)

*shkim@kinac.re.kr

According to Article 3(2) of the NPT and NSG the Guidelines, the exporting country should be guaranteed the import country's willingness to implement nuclear non-proliferation and the level of implementation before the transfer of Trigger List Items. Also, unknown or new end-users could be officially identified through GA procedure. Accordingly, Korea government requests the importing country a formal Governmental Assurance (GA), before issuing an export license. This study summarizes GA items and characteristics.

First, GA is The NSG guidelines suggests four items that should be assured by importing countries: peaceful use of export items, full-scope safeguards application, physical protection measures, and retransfer control. Therefore, these four items are generally requested based on the NSG guidelines. However, if they are already guaranteed by bilateral nuclear cooperation agreement, the GA could be based on the nuclear cooperation agreement. The GA procedure could be omitted in the case of concluding an administrative agreement that imposes another implementation procedure.

The levels of requested GA requested vary with the countries, since the NSG guidelines are just recommendations that are not legally enforceable. Korea requests the level based on the NSG guidelines. Among the four GA items, peaceful use of export items is the most fundamental obligation, and levels of safeguards and physical protection of the importing country could be verified in advance by reviewing the conclusion status of international agreement with the IAEA. Thus, the important thing for the licensee to consider is to decide the level of retransfer control.

The NSG guidelines suggest two levels of retransfer control, taking into account the sensitivity of the export items and the level of safeguards in the end-user country, which is either to receive GA from the third receiving country at an equivalent level with that originally requested from exporting country, or to get a prior consent by the exporting country. The latter should be approached more carefully, as requiring a prior consent is not only to have authority, but also to have responsibility when problem occurs. In addition, the level of retransfer control must be decided through sufficient reviewing on the transaction characteristics, it may affect the domestic export industry.

Keywords: Trigger List Items, Governmental Assurance (GA), NSG Guidelines, Bilateral Nuclear Cooperation Agreement, Retransfer Control

A Study on the License for Intermediate Trade and Merchandising Trade of Trigger List Items

Su-Hyeon Kim* and Han-Sol Ko

Korea Institution of Nuclear Nonproliferation And Control (KINAC)

*shkim@kinac.re.kr

Trades are classified as a direct trade, in which an exporter and an importer directly conclude a contract to execute a transaction, and an indirect trade, in which a transaction is conducted through a third party. A license issued by NSSC is required for the indirect trade of Trigger List Items although the items do not cross the Korea's borders. This study would summarize characteristics for each type of the indirect trade, and suggest things considered from the perspective of regulations.

Sensitive items such as weapons of mass destruction could be illegally transferred through black trade, and the role of brokers is important. Therefore, Korea controls indirect trade of Trigger List Items in a more conservative approach, although NSG guidelines does not include it as a control scope. Advanced countries in export control field such as the US, UK, and Canada also controls on indirect trade.

One of the indirect trade types is an intermediate trade in which the goods are imported from a foreign country and exported to another foreign country without being brought into the trader's country. Another type is a merchandising trade which is a form of brokerage between exporter from a foreign country and importer from another foreign country. Both types have one thing in common that the goods are not crossed the trader's country. The difference is that an intermediate trader directly participates in the contract and earn a difference between import and export amount, whereas a merchandising trader just arranges the transaction and earns a brokerage fee. The profits from the intermediate trade are considered as export records, while merchandising trade profits are not considered export records. In other words, only the intermediary trade is considered as an export. Also, the license types are different for each of them. An export license should be issued for the intermediate trade of Trigger List Items, whereas a brokerage license should be issued for the merchandising trade of Trigger List Items.

The definition of export in the Foreign Trade Act includes intermediate trade for only goods, but technology is missing, although the sub-regulation specifies the intermediate trade including both. The technology need to be added as it can be the subject of intermediate trade in spite of intangible characteristics. Also, outreach activities are more needed as nuclear industry awareness on export control for both trades is low.

Keywords: Trigger List Items, Intermediate Trade, Merchandising Trade, Export License, Brokering License

Considerations for Nuclear Power Plant Cyber Security Contingency Plan

Seungmin Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*smkim90@kinac.re.kr

Nuclear power plants (NPPs) are designed in consideration of redundancy, diversity, and independence to prevent leakage of radioactive materials from safety of view, and a contingency plan is established in case of DBA (Design Basis Accident) occurrence. In addition, NPPs have established contingency plans for physical attacks, including terrorist intrusions and bomb attacks. However, the level of contingency plan caused by cyberattacks is quite insufficient compared to the contingency plan in terms of safety and physical protection. The purpose of this paper is to present the problems of cyberattack contingency plan and methods to supplement it.

The first problem with cyberattack contingency plan is that the initiating event for implementing the contingency plan is undecided. In terms of safety, the DBA is identified as an initial event, and each contingency plan is based on the initial events specified in the DBA such as Loss of Coolant Accident and Loss of Offsite Power. In terms of physical protection, each has a contingency plan by identifying bomb attacks and terrorist intrusions in Protected Area and Vital Area as initial events. On the other hand, in the contingency plan related to a cyberattack, an initial event caused by a cyberattack is not identified. For this, it is necessary to classify the attack results that may occur when the CDA is compromised based on the attack technique described in Design Basis Threat. Based on this, an initiating event should be selected and a contingency plan according to each initiating event should be established.

The second problem is that there is no responsibility matrix according to the occurrence of the initiating event. From a safety point of view, when a DBA occurs, the organization's mission according to each initial event is described in the contingency plan, and related countermeasures are defined in case of an accident through Emergency Operation Procedure. In the case of physical protection, referring to IAEA's Regulatory Guide 5.54, the organization's responsibility is defined in matrix form when an initial event such as a bomb attack occurs. In this way, the responsibility matrix to be carried out in case of initiating events based on the defined initial event should be described in the contingency plan.

In this paper, the problems of the cyberattack contingency plan are presented, and for this purpose, the definition of the initial event and the need for a responsibility matrix when the initial event occurs are presented.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1804025).

Keywords: Cyber Security, Contingency Plan, Initiating Event, Responsibility Matrix

Estimation of U Ore Production Using Bayesian Network

Yonhong Jeong* and Hojung Do

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*jyh1404@kinac.re.kr

Concerns about North Korea's 7th nuclear test have been rising recently, and it is a significant threat to the situation around the Korean Peninsula. Amidst these threats, the Korean government also shows a strong will for denuclearizing the Korean Peninsula, referring to the "Audacious Initiative." For denuclearization negotiations with North Korea, it is essential first to understand North Korea's nuclear capabilities. However, since access to information is complicated and contains many uncertainties, many studies have been conducted to estimate it. Among them, Von Hippel surveyed to estimate the total amount of uranium ore based on information on uranium mining, which is relatively widely known throughout North Korea's nuclear fuel cycle, and the amounts of HEU and Pu suggested by many experts. KINAC has conducted a study on a methodology that can narrow the estimation range and improve reliability through the Bayesian Network based on Von Hippel's research results. However, in this study, the probability distribution is assumed to be the simplest form of uniform distribution, and the estimation formula for the amount of Pu produced compared to the amount of uranium loaded in the core is used as it is, which is an error in Von Hippel's study. Improvement is needed. This study proposes a more reliable BN model by supplementing this and attempts to estimate the amount of uranium ore that North Korea produces or possesses. Of course, the data used as the basic structure of the model is insufficient, and the estimation formula is straightforward, so it is somewhat unreliable to trust the estimate for uranium ore. However, it is expected to be a suitable methodology that can narrow the scope of North Korea's nuclear material production estimate or compensate for the uncertainty of the nuclear material production estimation model being developed at KINAC.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1905008-0119).

Keywords: Nuclear non-proliferation, Denuclearization, Bayesian Network, U Ore Production

2분과

사용후핵연료 처분전관리(Oral)

Machine Learning Based Quality Control for Spent Fuel Safety Information

Ga-hee Sim, Moon-Ghu Park*, and Kyoong Ho Cha

Sejong University

*mgpark@sejong.ac.kr

The spent fuel safety information delivered from the consignor to the disposal facility operator directly affects the operation and safety of the disposal facility. Therefore, the operator of a disposal facility must perform data quality management to increase data reliability, and anomaly detection is a representative method among quality control methods. We propose a quality control method to detect anomalies using XGBoost, known for its excellent performance, prevention of overfitting, and fast training speed. First, we select significant variables such as release burnup, enrichment, and amount U from the spent fuel safety information and train models for each variable using only normal data. A model trained using only normal data generates a small error for a normal pattern and a large error for an abnormal pattern. Then, when the data error exceeds a set threshold, the data is determined as an anomaly.

In this paper, we implement the XGBoost models using virtual spent fuel information and optimize the hyperparameter of XGBoost using a simulated annealing method for high accuracy. The optimized XGBoost models show high accuracy in a normal input and provide a stable prediction value even in an abnormal input. In addition, we perform anomaly detection by including defect input in the data to validate the presented method. The proposed method shows the result of effectively classifying normal values and anomalies.

Keywords: Machine Learning, XGBoost, Quality Control

Electrochemical Impedance Spectroscopy Study of Hastelloy N in Molten LiCl-KCl Salt at 500°C

Jun Woo Park and Jong-Il Yun*

Korea Advanced Institute of Science and Technology (KAIST)

*jiyun@kaist.ac.kr

One of the promising candidates for heat transfer fluid is molten chloride salts. They have been studied in various fields such as the electrolyte of pyroprocessing, the molten salt reactor coolant, and the energy storage system media. Main considerations for utilizing molten chloride salts are the compatibility of salts with structural materials. The corrosion behavior of structural materials in molten chloride salts must be understood to identify suitable materials against the corrosive environment.

In this study, the corrosion behavior of a candidate structural material, Hastelloy N, in molten LiCl-KCl salt at 500°C were investigated by the electrochemical impedance spectroscopy (EIS) method. The sheet type of Hastelloy N was utilized as the working electrode in LiCl-KCl to measure the EIS data for 100 hours with 5 hours of time intervals. The EIS data were measured in the frequency range from 10^4 Hz to 10^{-2} Hz with the AC signal (amplitude = 20 mV) at open circuit potential.

The capacitance semicircle observed in Nyquist plots for all periods indicates that charge-transfer controlled reactions occur. As the immersion time progresses, the radius of the semicircle in Nyquist plots and the impedance and phase angle in Bode plots decrease. These behaviors suggest a decreasing reaction resistance and the corrosion reactions are accelerated with the immersion time. The EIS data were fitted using the equivalent circuit to achieve quantitative results. Two capacitor-resistor components were considered due to the overlapped shape of two valleys in phase angle. The depressed shape of the semicircle in Nyquist plots led to the use of the constant phase element(Q) instead of the capacitor. Therefore, $R(Q(R(QR)))$ circuit was selected to fit the EIS data. Fitting results show that the charge transfer resistance decreases dramatically within 1 day and then converges. The film resistance shows no clear trends, but the increase of the film admittance value indicates the decreased film thickness. Consequently, the film appears to exist like the oxide layer but it does not act as a protective layer.

The real-time EIS data were measured in molten salt and provides the corrosion behavior over time. The corrosion mitigation strategy should consider that the corrosion of Hastelloy N accelerates over time and its intrinsic film cannot act as the protective layer. The next steps of this study are to evaluate other candidate structural materials and to demonstrate the presence of the film.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP: Ministry of Science, ICT and Future Planning) (No. NRF-2020M2D4A1070724 and NRF-2021M2C7A1A02076339) and the Circle Foundation (Republic of Korea) as the ‘2021 TCF Innovative Science Project’.

Keywords: Molten Salt, Structural Material, Electrochemistry, Corrosion, Electrochemical Impedance Spectroscopy

Separation of Sr From Simulated Oxide Fuel Using Molten Chloride Salts

Eun-Young Choi^{1,2,*}, Sang-Kwon Lee¹, Seungwoo Paek¹, Hamin Choi¹, Minsu Lee², DongHyun Jeon¹,
and Change Hwa Lee¹

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*eychoi@kaeri.re.kr

Separation of high heat generating-radioactive isotopes from spent nuclear fuel is an important issue because it can reduce the final disposal area. As one of the technologies that can selectively separate only high heat generating-radioactive isotopes without dissolving spent fuel, the methods using molten salt have recently attracted attention. Although studies on chemical changes of Sr oxides in molten salts have been reported, they have limitation in that alternative oxide reagents rather than oxide fuel were used. In this study, the separation behaviors of Sr from simulated oxide fuel using various molten salts were investigated. A powder type containing 95.7wt% of U and 0.123wt% of Sr was used as the simulated oxide fuel. LiCl, LiCl-CaCl₂, MgCl₂, LiCl-KCl-MgCl₂ and NaCl-MgCl₂ were used as molten chloride salts. The separation of Sr from the simulated oxide fuel was conducted by loading it in porous alumina basket and immersing it in a salt. The concentration of Sr in the salt was measured by ICP analysis after sampling the salt outside the basket using dip-stick technique. The separation efficiencies of Sr from simulated oxide fuel using the salts were compared. Furthermore, the causes of their separation efficiency were systematically investigated.

Keywords: Molten Salt, Simulated Oxide Fuel, Radioactive Isotopes, Strontium

Investigation of Radial Hydride Effect on Mechanical Properties of Zircaloy Cladding Via Hydride Morphology Image Analysis

Dahyeon Woo and Youho Lee*

Seoul National University (SNU)

*leeyouho@snu.ac.kr

Hydride reorientation is one of the major concerns for cladding integrity during dry storage. In this study, mechanical property of post-reorientation cladding was investigated according to the morphology and amount of the hydrides. Cladding peak temperature limit 400°C was suggested by U.S. NRC in concern of cladding creep and hydride reorientation. In line with this regulatory limit, hydride reorientation was conducted during cool-down process from the maximum temperature of 400°C, using constant internal pressurization method. The specimens were charged for hydrogen from 100 to 1,000 wppm, and various pressures range of 7.5-18.5 MPa were applied. The morphology was examined by optical microscopy. Radial hydride fraction (RHF) and radial hydride continuous path (RHCP) were calculated using image analysis software PROPHET. Finally, strain energy density (SED) was investigated via ring compress tests and the hydrogen concentration was analyzed. The result shows that when RHF is higher than 5%, SED exponentially decreases with RHF. For RHF less than 5%, SED was primarily affected by the total amount of hydrogen. Shortened length of radial hydrides with the presence of circumferential hydrides may block the radial propagation of crack. The result implies that lower burnup spent fuel with lower hydrogen concentration may be more vulnerable in terms of radial hydride compared to higher burnup fuel.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Ministry of Trade, Industry and Energy (MOTIE) using the financial resource granted by the Institute for Korea Spent Nuclear Fuel (iKSNF) of the Republic of Korea [No.2021040101002A].

Keywords: Hydride Reorientation, Radial Hydride Fraction, Radial Hydride Continuous Path, Spent Nuclear Fuel, Dry Storage

A Study on the Properties of B₄C/Al Composites for Absorbing Thermal Neutrons Manufactured by the Stir Casting Process

Donghyun Lee, Seungchan Cho*, Junghwan Kim, Sang-bok Lee, and Sang-kwan Lee

Korea Institute of Materials Science

*sccho@kims.re.kr

B₄C/Al composite is mainly used for neutron absorbing materials, which is one of the components of equipment that manages spent nuclear fuel.

There are various processes for manufacturing neutron absorbing materials, but most of them are based on the powder metallurgy.

In this study, B₄C/Al composite in which the reinforcement was uniformly dispersed was manufactured by using the stir casting process.

The microstructure, thermal neutron absorption rate, mechanical properties and dispersibility of the reinforcement of the prepared B₄C/Al composite were analyzed.

Keywords: MMC, Stir Casting, Thermal Neutron Shielding, Boron Carbide

Evaluation of Physicochemical Characteristics of Group II Nuclide Ceramic Waste Form for Long-term Storage and Disposal

Byeonggwon Lee^{1,2}, Jung-Hoon Choi^{2,*}, Ki Rak Lee², Hyun Woo Kang², and Hwan-Seo Park²

¹Kyungpook National University (KNU)

²Korea Atomic Energy Research Institute (KAERI)

*mrchoijh@kaeri.re.kr

To minimize the short-term thermal load on the repository facility, heat generating nuclides such as Cs-137 and Sr-90 should be separated from the spent nuclear fuel for efficiency of repository facility. In particular, Sr-90 must be separated because it generates high heat during the decay process. Recently, Korea Atomic Energy Research Institute (KAERI) is developing a waste burden minimization technology to reduce the environmental burden caused by the disposal of spent nuclear fuel and maximize the utilization of the disposal facility. The technology includes a nuclide management process that can maximize disposal efficiency by selectively separating and collecting major nuclides such as Cs, Sr, I, TRU/RE, and Tc/Se from spent nuclear fuel. Among the major nuclides, Sr nuclides dissolve in chloride phase during the chlorination process of spent nuclear fuel and recovered in the form of carbonate or oxide via reactive distillation. In this process, Ba nuclides are also recovered along with Sr nuclides due to their chemical similarity. In this study, we prepared group II nuclide ceramic waste form, $\text{Ba}_{(x)}\text{Sr}_{(1-x)}\text{TiO}_3$ ($x=0, 0.25, 0.5, 0.75, 1$), using the solid-state reaction method by considering the various ratio of Sr/Ba nuclides generated from nuclide management process. The established waste form fabrication process was able to produce a stable waste form regardless of the ratio of Sr/Ba nuclides. To evaluate the stability of group II waste form, physicochemical properties such as leaching and thermal properties were evaluated. Also, the radiological properties of the $\text{Ba}_{(x)}\text{Sr}_{(1-x)}\text{TiO}_3$ waste forms with various Sr/Ba ratios were evaluated, and the estimation of centerline temperature was carried out using the experimental thermal property data. These results provided fundamental data for long-term storage and management of group II nuclides waste form.

Keywords: Strontium-90, Group II Nuclides, Ceramic Waste Form, Centerline Temperature, Nuclide Management Process

An Analysis of Neutron and Gamma Sources in Spent Fuels Using SCALE-ORIGEN ARP

Sohee Cha and Kwangheon Park*

Kyung Hee University

*kpark@khu.ac.kr

The management before disposal of spent nuclear fuel is an essential process for safe management. It is important to determine the amount of nuclide inventory in order to ensure the integrity of spent nuclear fuel, as radiation generated from the nuclides is generated along with residual heat in the spent nuclear fuel. Based on the data on the characteristics of spent nuclear fuel generated in Korea, the correlation equation between burnup and enrichment was derived by referring to overseas cases (Sweden). Source term analysis was performed using the SCALE ORIGEN ARP code by securing the burnup history of nuclear fuel. Calculation was performed by inputting the combustion history of the fuel WH14×14 and WH17×17 as a reference for CE16×16 spent fuel. Through this study, the relationship was identified using the burnup, enrichment, and cooling time factors that influence the characteristics of spent nuclear fuel. In addition, the total source and spectrum data from neutrons and gamma sources were used to find out the characteristics of fuel.

Acknowledgements

This work was partly supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (20222B10100060, Development of On-site Burn-up Detection System for the Spent Fuel).

Keywords: Nuclear Spent Fuel, Source Term, Spectrum, Burnup, Cooling Time, Enrichment

Development of Model to Evaluate Thermal Fluid Flow Around a Submerged Transport Cask of Spent Nuclear Fuel in the Deep Sea

Guhyun Jeoung, Sungyeoun Kim, and Sanghoon Lee*

Keimyong University

*shlee1222@gmail.com

As the saturation rate of temporary storage facilities for spent nuclear fuel increases, regulatory demands such as interim storage and permanent disposal of spent nuclear fuel are expected to begin in earnest. Considering the domestic situation where all nuclear power plants are located on the waterfront site, the interim storage site is also likely to be located on the waterfront site, and maritime transportation is one of the essential management stages. Currently, there are no independently developed maritime transportation risk assessment code in Korea, and no research has been conducted to evaluate the release of radioactive waste due to the sinking of transport container. Therefore, it is necessary to secure technology to properly reflect the domestic maritime transportation environment and to assess the impact of the sinking accident and to carry out safety regulations. To accurately calculate the releaser rate of radionuclides contained in a cask with breached containment boundary, the flow rate through the gap generated in the containment boundary should be calculated. The fluid flow through this gap which is probably in micro scale in most situations should be evaluated combining the fluid flow inside and outside the cask. In this study, a detailed computational fluid dynamics model to evaluate the internal fluid flow in the cask and a simplified model to capture the fluid flow and the heat transfer around the cask in the sea are constructed. The results for the large scale model are compared with the analytic formula for verification of heat transfer coefficient and they showed good agreements. The heat transfer coefficient thus found can be used in the detailed model to provide more realistic data than those obtained from assumed heat transfer coefficient around the surface of the cask. In the future, fluid flow through the gap between the lid and the body of the cask will be evaluated coupling the models developed in this work.

Keywords: Spent Nuclear Fuel, Transport Container, Maritime Transportation, Sinking Accident, Computational Fluid Dynamics Model, Simplified Model

2분과

사용후핵연료 처분전관리(Poster)

Spent Fuel Characteristics of Small Modular Reactors and Its Management Implications

Dahyeon Woo, Chansoo Lee, and Youho Lee*

Seoul National University (SNU)

*leeyouho@snu.ac.kr

Since SMR's reduced reactor radius results in higher neutron leakage, SMR operates at a relatively lower discharge burnup level than traditional Light Water Reactors (LWRs). It may result in larger spent fuel amounts for SMRs. Furthermore, recent studies demonstrated that NuScale reactor will generate a significantly higher volume of low- and intermediate-level waste owing to components located near the active core including the core barrel and the neutron reflector. For spent nuclear fuel simulation, FRAPCON-4.0 was updated. Major modifications were made for fission and decay gas release, pellet swelling, cladding creep, axial temperature distribution, corrosion, and extended simulation time covering from steady-state to dry storage. In this study, typical 17×17 PWR fuel (60 MWd/kgU) and NuScale Power Module (36 MWd/kgU) was compared. NuFuel-HTP2™ fuel assembly, which has a half-length of proven LWR fuel, was employed. Owing to the lower discharge burnup and operating temperature, the maximum hydrogen pickup was 73 wppm and the maximum hoop stress was ~25 MPa. Therefore, hydride reorientation issue is irrelevant to SMR spent fuel. In this context, the current regulatory limit for dry storage (i.e. 400°C and 90 MPa) can be significantly alleviated for LWR-based SMRs. The increased safety margin for SMR spent fuel may compensate high spent fuel management cost of SMRs incurred by an increased amount. The comprehensive analysis on SMR spent fuel management implications are discussed based on simulated SMR fuel characteristics.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the institute for Korea Spent Nuclear Fuel (iKSNF) using the financial resource granted by the Ministry of Science and ICT (MSIT) of the Republic of Korea [No. 2021M2E1A108598612].

Keywords: Small Modular Reactor, Spent Nuclear Fuel, Dry Storage

Analysis of Hydrogen Migration in Zircaloy Cladding During Dry Storage by Simulating Hydrogen Diffusion and Phase Change Kinetics

Chansoo Lee and Youho Lee*

Seoul National University (SNU)

*leeyouho@snu.ac.kr

This study reassess safety margin of the current Peak Cladding Temperature (PCT) limit of dry storage in terms of hydrogen migration by predicting axial hydrogen diffusion throughout dry storage with respect to wet storage time and average burnup. Applying the hydride nucleation, growth, and dissolution model, an axial finite difference method code for thermal diffusion of hydrogen in zirconium alloy was developed and validated against past experiments. The developed model has been implemented in GIFT – a nuclear fuel analysis code developed by Seoul National University. Various discharge burnups and wet storage time relevant to spent fuel characteristics of Korea were simulated. The result shows that the amount of hydrogen migrated towards the axial end during dry storage for reference PWR spent fuel is limited to ~50 wppm. This result demonstrates that the current PCT margin is sufficient in terms of hydrogen migration.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the institute for Korea Spent Nuclear Fuel (iKSNF) using the financial resource granted by the Ministry of Science and ICT (MSIT) of the Republic of Korea [No. 2021M2E1A108598612].

Keywords: Hydrogen Diffusion, Hydride, Zirconium Alloy, Dry Storage

Development of a Radial Hydride Fraction Prediction Model and Experimental Validation

Changhyun Jo and Youho Lee*

Seoul National University (SNU)

*leeyouho@snu.ac.kr

As the zircaloy cladding absorbs an excessive amount of hydrogen and cooled down under hoop stress, radial hydride may be precipitated by hydride reorientation phenomenon. There have been many previous studies about the threshold stress of the reorientation, but it is known that the quantitative degree of hydride reorientation rather than the threshold is important for the prediction of mechanical properties. A thermodynamic model for Radial Hydride Fraction (RHF) prediction has been developed in this study. The model calculates RHF with respect to temperature, cooling rate, hydrogen content, and applied stresses. Once the cooling rate is given, the solid solution concentration at each temperature is determined by Hydrogen-Nucleation-Growth-Dissolution model. Subsequently, the increment of radial hydride is derived by nucleation and growth theory. The code based on the thermodynamic theory can provide the prediction of RHF under hoop stress, as well as a change in precipitation behavior over time. RHF of the zircaloy cladding in long-term dry storage can be obtained by the implementation of the code and the degradation of the cladding is directly estimated according to the correlation between RHF and mechanical properties. Ongoing experimental validation of the developed model is discussed.

Acknowledgements

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

Keywords: Hydride Reorientation, RHF, Dry Storage, Thermodynamic Modeling

Comparison of Fire Test Using Slice Model and Half-scale Model of the KTC-360 Transport Cask

Kyoung-sik Bang*, Yun-Young Yang, and Woo-Seok Choi

Korea Atomic Energy Research Institute (KAERI)

*nksbang@kaeri.re.kr

According to the ‘Basic Plan for High-Level Radioactive Waste Management (draft)’, the total amount of CANDU spent nuclear fuel is expected to be approximately 660,000 bundles. To safely and efficiently transport this amount to interim storage facilities, it is essential to develop a large-capacity transport cask. Therefore, we have been developing a large-capacity PHWR spent nuclear fuel transport cask, called the KTC-360 transport cask. According to the transport-cask related regulations, the KTC-360 transport cask was classified as a Type B package, and such packages must be able to withstand a temperature of 800°C for a period of 30 min. It is desirable to conduct a test using a full-scale model of a shipping package when performing tests to evaluate its integrity. However, it is costly to perform a test using a full-scale model. Therefore, to evaluate the thermal integrity of the KTC-360 transport cask, the fire test was conducted using a slice model. For comparison purposes, the fire test was also carried out using a 1/4 scale model. In the fire test using a slice model and in the fire test using a 1/4 scale model, the maximum temperature of the cask body was lower than the permitted maximum temperature limit. Therefore, the thermal integrity of the KTC-360 transport cask could be considered to be maintained. The temperature results from the fire test using a slice model were higher than those of the fire test using a 1/4 scale model. Therefore, the effect of flame on a transport cask without combustible materials, such as the KTC-360 transport cask, seems to be affected by the reduction in the time rather than the size reduction.

Acknowledgments

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and Ministry of Trade, Industry and Energy (MOTIE) of the Republic of Korea (No. 2020 1710200010).

Keywords: Pressurized Heavy Water Reactor, Fire Test, Thermal Integrity, Transport Cask

Suggestion of Simple Rod Internal Pressure and Free Void Volume Correlations in Spent Fuel Rod

Yongsik Yang* and Donghak Kook

Korea Atomic Energy Research Institute (KAERI)

*yys@kaeri.re.kr

A rod internal pressure increased by fission gas release is major factor that causes degradation during dry storage of spent fuel. Because rod internal pressure is greatly affected by fuel design, operation power history, it is essential to perform complex calculation using performance code to accurately predict rod internal pressure as function of burnup.

However, because it is difficult to apply a complex method into dry storage design and to determine rod internal pressure based on conservative way this study presents a simple correlation that can predict an approximate rod internal pressure as function of burnup. For the development of simple correlation, rod internal pressure and fuel rod void volume data measured through about 400 PIE (Post Irradiation Examination) data were used. The developed simple correlation can cover various fuel rod arrays, discharged fuel average burnup, operation history, cladding type, burnup range, and information on Westinghouse type fuel rods such as Spain ENUSA, USA EPRI/ANL/ORNL/PNNL, WEC, etc.

In this paper, the data of simple correlation determination is briefly introduced, and the data analysis process and results are summarized. Two correlations that can conservatively determine rod internal pressure and free void volume in fuel rod according to fuel rod average burnup were presented, and the effect of initial He fill pressure was evaluated.

In particular, the results of Post Irradiation Examination for 46 fuel rods conducted in Korea are also included, so it is expected that newly presented correlations can be used easily in various ways in the domestic research, industry, and academia.

Keywords: Spent Fuel, Rod Internal Pressure, Rod Void Volume, Fuel Rod, Fuel Performance

Analysis of the Regulatory Processes for Spent Nuclear Fuel Transportation Cask

Hyun Jun Na, Hyeok Jae Kim, Min Seong Kim, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

For safe management of spent nuclear fuels, they should be delivered to repository or waste disposal site. As the amount of spent nuclear fuel transportation is expected to increase in the future due to the provision of an intermediate storage facility, the necessity to secure transportation cask is emerging. In order to secure the spent nuclear fuel transportation cask, it is necessary to analyze the regulatory processes for domestic and foreign spent nuclear fuel transportation cask. In this study, the regulatory processes for domestic and foreign spent nuclear fuel transportation cask was analyzed.

In this study, the IAEA, US, and Korea spent nuclear fuel transportation cask regulatory processes were analyzed. The domestic and foreign spent nuclear fuel transportation cask regulatory processes consist of design phase, manufacturing phase, and operation phase. In the design stage, the transport requirements are designed in accordance with the safety requirements of international organizations and countries. The application to be submitted when applying for approval should include a safety analysis report, evidence proving compliance with safety requirements et al. In the manufacturing stage, it is a stage to check whether the safety requirements are satisfied before the first use after manufacturing the transportation cask. Inspections include welding inspection, leakage inspection, shielding inspection, and thermal inspection. In the operation stage, it is a stage of periodically performing inspections for continuous maintenance of the package when the transportation cask is used. The inspection items to be performed are similar to the manufacturing stage and typically include performance inspection of components and leakage inspection.

In this study, domestic and foreign spent nuclear fuel transportation cask regulatory processes were analyzed. It was found that the domestic and foreign spent nuclear fuel transportation cask regulatory processes consist of the design phase, the manufacturing phase, and the operation phase. The results of this study can be used as basic data for policy decision-making for the spent nuclear fuel cask.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2021 171020001B).

Keywords: Spent Nuclear Fuel, Transportation Cask, Regulatory Process, Safe Management

Decay Heat Estimation of Spent Fuel Assemblies for Decommissioning of Kori Units 3 and 4

Wonjune Mah*, Chang-Lak Kim, and Sun-kee Lee

Kepco International Nuclear Graduate School (KINGS)

*wjmah@email.kings.ac.kr

This paper mainly focuses on the maximum decay heat estimation generated from spent fuel assemblies in the spent fuel pool of Kori units 3&4 at the beginning decommissioning.

It is assumed that the spent fuel pool is fully occupied with 2,260 spent fuel assemblies, same as its design capacity. In addition, equally 56.5 spent fuel assemblies have been generated per year. The minimum cooling time is five years considering the transition phase between the permanent shut-down and the amendment of Operating License for decommissioning. Sending and receiving of spent fuel assemblies to/from other units are neglected.

Seven representative spent fuel assembly groups are established based on the burnup rate and cooling time. Conservatively high values for the burnup rates and low values for the cooling times are applied. Calculation of the decay heat of each representative group has been performed by using ORIGEN decay solver of SCALE. Then, total decay heat has been calculated based on this.

Group 1, 2, and 3 contain comparatively old spent fuel assemblies with 45 GWd/tU burnup rate and 20~30 cooling years. The calculation shows 489~586 watts of decay heat per assembly. Group 4, 5, 6, and 7 contain comparatively new spent fuel assemblies with 55 GWd/tU burnup rate and 5~20 cooling years. The calculation shows 741~1,483 watts of decay heat per assembly. The total maximum decay heat therefore is estimated as 1,609,459 watts.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP Project No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Spent Fuel Assembly, Spent Fuel Pool, Decay Heat, Decommissioning, Kori Nuclear Power Plant

Examination for Evaluating Degradation of Guide Tubes in Spent Fuel Assembly

Myunchul Park*, Sunggeun Kim, Jaehan Kim, Sukwoo Hong, and Hyojik Lee

Korea Atomic Energy Research Institute (KAERI)

*mcpark@kaeri.re.kr

The skeleton of fuel assembly is composed of top nozzle, bottom nozzle, grids, and guide tubes. In the reactor core, all the parts of the fuel assembly suffer degradations due to the condition of high temperature, pressure and water environment. Therefore, many material properties of high temperature mechanical strength, corrosion and irradiation resistance have been considered to choose the material for fuel assembly parts in the fuel development stage. The guide tubes have important roles to connect each parts and support the load of fuel assembly while the fuel is lifted. In Westinghouse 14×14 standard fuel assembly, Zircaloy-4 was used for the material of the guide tubes. Zircaloy-4 has a resistance to water corrosion and maintain good mechanical properties after the discharge from the core, so this alloy is also utilized for a fuel rod cladding material although the microstructure is slightly different due to the heat treatment difference. Thus, it is expected that there is no issue regarding the guide tube integrity after the discharge and during the storage in the pool, especially in case of low burn-up. However, the surface oxidation and resultant hydrogen pick-up can affect to the embrittlement to the Zr alloy. So, it is needed to know the actual status of spent fuel assembly by performing post-irradiation examination. In this study, the degradation level of the guide Tubes in low burn-up spent fuel assembly was investigated using the KAERI PIE facility in order to make some data which can be utilized to the baseline for evaluating the integrity of the spent fuel skeleton.

Keywords: Guide Tube, Zircaloy-4, Hydrogen, Oxidation, PWR Spent Fuel Assembly

Introduction of Spent Fuel Information Management Model Based on Semantic Web Technology

Hong-jin Kim, Young-Chul Cho, and Kyoong-Ho Cha*

Sejong University

*khcha@sejong.ac.kr

Numerous spent nuclear fuels are generated every year in Korea. To solve the spent nuclear fuel problem within saturated temporary storage, the authorities are readying to build an interim storage and a permanent disposal facility in the country. At the same time, the authorities are readying to establish a management procedure for spent nuclear fuel. In the future, the authorities need to make and apply the Database of spent nuclear fuel to practice the management procedure. However, the structure of a traditional database is not reasonable for information management because it has a problem with listing data and identifying data features due to its structure. In addition, the traditional database always exists human error from working in Excel program by a human. Therefore, this research proposes a new standard information management model based on Semantic Web technique. Semantic Web uses a data structure named ontology. By using the ontology in the information database of the spent nuclear fuel, users, such as institutions related to management, could more easily recognize and understand the Database. Furthermore, since this task proceeds in the ontology construction program, the human error in the new model reduces rather than an environment of the traditional database.

Keywords: Spent Nuclear Fuel, Information Management, Database, Structure, Semantic Web, Ontology

Effect of Li_2O Addition on Precipitation Behavior of Uranium in LiCl-KCl-UCl_3 Salt

Taeho Kim*, Su Jeong Heo, Dalsung Yoon, Seungwoo Paek, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*tkim@kaeri.re.kr

The effect of Li_2O addition on precipitation behavior of uranium in LiCl-KCl-UCl_3 has been investigated in this study. 99.99% LiCl-KCl eutectic salt is mixed with 10wt% UCl_3 chips at 550°C in the Pyrex tube in argon atmosphere glove box, with 10 ppm O_2 and 1 ppm H_2O . Then, Li_2O chunks are added in mixed LiCl-KCl-UCl_3 and the system has been cooled down to room temperature for 10 hours to form enough UO_2 particles in the salt. The solid salt has been taken out from the glove box, and cut into three sections (top, middle and bottom) by low-speed saw for further microscopic analysis. Three pieces of solid salt are dissolved in deionized water at room temperature and the solution is filtered by a filter paper to collect non-dissolved particles. The filter paper with particles is baked in vacuum oven at 120°C for 6 hours to evaporate remaining moisture from the filter paper. Further analysis was performed for the powder remaining on the filter paper, and periphery of the powder (cake) on the filter paper. Scanning electron microscopy (SEM), electron diffraction spectroscopy (EDS), and X-ray powder diffraction (XRD) are adopted to analysis the characteristic of the particles. From SEM analysis, the powders are consisted of small particles which have 5 to 10 μm diameter, and EDS analysis shows they are likely UO_2 with 23 at. % of uranium and 77 at. % oxygen. Cake is also analyzed by SEM and EDS, and needle like structures are widely observed on the particle. The length of needle is distributed from 5 to 20 μm , and it has 6 to 10 at. % of chlorine, which are not fully dissolved into deionized water at room temperature. From XRD analysis, the particles show the peak position of UO_2 , and the result is well matched with the SEM-EDS results. We are planning to add more Li_2O in the system for fully reacting uranium in UCl_3 , and compare the results to find the effect of Li_2O concentration on UO_2 precipitation.

Keywords: Li_2O Addition, Uranium Precipitation, UO_2 , LiCl-KCl-UCl_3 Salt

Separation of Salt From Strontium Carbonate and Uranium Oxide Powders

Sang Woon Kwon*, Su Jeong Heo, Joon Bo Shim, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*swkwon@kaeri.re.kr

It has been studied on the disposal area reduction for the used nuclear fuel by the management of high decay-heat nuclides, long-lived nuclides, and highly mobile nuclides. It was investigated on the management of the nuclides in KAERI. Strontium-90 is a high heat-generating nuclide in spent nuclear fuel. It is needed to separate the salt from the salt solution for the recovery of strontium after the chlorination of the strontium oxide in molten salt. Vacuum distillation was used for the separation of strontium from the molten salt. Potassium carbonate was chosen as a reactive distillation reagent for $\text{SrCl}_2 - \text{LiCl} - \text{KCl}$ system by the thermodynamic calculation. Reactive distillation experiments were carried out. The residual was mainly SrCO_3 in the XRD analysis. It could be concluded that K_2CO_3 could be one of the suitable reagents for the reactive distillation. The salt in the long-lived nuclide powders should be removed to prepare the block for disposal. Experiments were carried out using W powders (surrogate) and U_3O_8 powders to develop a process for the removal of the residual salt from UOx powders. The salts were successfully removed from the W and U_3O_8 powders by distillation.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MEST) (No. 2021M2E3A1040059).

Keywords: Strontium, Chlorination, Recovery, Molten Salt

UO₂ and SIMFUEL Pellets Fabrication and Leaching Behavior Study for Stabilization Technology of Damaged Used Fuel

Su Jeong Heo*, Ju Ho Lee, Jae-Won Lee, and Yung-Zun Cho

Korea Atomic Energy Research Institute (KAERI)

*sujeongheo@kaeri.re.kr

Irradiated uranium dioxide in damaged used fuel could oxidize during transportation, interim storage or disposal, resulting that the fuel pellet fragments are reduced to a grain-sized powder that can easily escaped from the damaged rod. It has been reported that oxidized spent fuel (i.e. U₄O_{9+x}) that was in contact with water could increase the dissolution rate by making the grain boundaries more accessible to the water. Therefore, the damaged used fuel requires stabilization technology including nuclear material recovery, pellet manufacturing process, and stabilization fuel rod manufacturing that can secure safety in terms of permanent disposal. In this study, we prepared pure UO₂ and SIMFUEL pellets that are a mixture of UO₂ and surrogated metallic oxides for fission products equivalent to a burn-up of 35 GWd/tU and 55 GWd/tU as the stabilized spent fuel. The UO₂ and fission products powders were milled and pressed into pellets at 250 MPa and sintered at 1,550°C and 1,700°C for 6 hours in an atmosphere of 4%H₂-Ar. The prepared UO₂ and SIMFUEL pellets were placed in PTFE Teflon vessels and filled with deionized water to identify the leaching behavior by a long-term leaching experiment under the similar condition to a repository for the safe disposal.

Keywords: Irradiated Uranium Dioxide, Spent Fuel, SIMFUEL, Leaching, Sintering

A Study on Optimization of UCl_3 Production by Reaction of Uranium Metal and Ammonium Chloride

Dalsung Yoon*, Seungwoo Paek, Sang-Kwon Lee, Ju Ho Lee, Taeho Kim, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*dyoon@kaeri.re.kr

Interests in molten salt reactor (MSR) using a fast spectrum (FS) have been increased not only for having a high power density but for burning the high-level waste generated from nuclear power plants. For developing the FS-MSR technologies, chloride-based fuels are considered due to the advantage of higher solubility of actinides and lanthanides over fluoride-based salts. Despite significant progress in development of MSR technology, the manufacturing technology for production of the fuel is still insufficiently understood. One of the option to prepare the MSR fuel is to use products from pyroprocessing where oxide form of spent nuclear fuel is reduced into metal form and useful elements can be collected via electrochemical methods in molten salt system at high temperature. In order to chlorinate the products into chloride form, previous study used NH_4Cl to chlorinate U metal into UCl_3 in an airtight reactor. It was found that the U metal was completely chlorinated into chloride forms; however, impurities generated by the reaction of NH_4Cl and reactor wall were found in the product.

Therefore, in this work, the air tight reactor was re-designed to avoid the reaction of reactor wall by insertion of Al_2O_3 crucible inside of the reactor. In addition, the reactor size was increased to produce UCl_3 over 100 g. Using the newly designed reactor, U metal chlorination experiments using NH_4Cl chlorinating agent were performed to confirm the optimal experimental conditions. The detailed results will be further discussed.

Keywords: Molten Salt Reactor, Molten Fuel Fabrication, Chlorination, Uranium Chloride

Fluidization Bed Behaviors of Uranium Oxide Powder Using Surrogates

Jae Won Lee*, Ju Ho Lee, Sun Seok Hong, and Yung zun Cho

Korea Atomic Energy Research Institute (KAERI)

*njwlee@kaeri.re.kr

The damaged spent fuel rods must be stabilized by encapsulation or dry re-fabrication technologies before geological disposal. For applying the dry re-fabrication technology, we manufactured a vertical type furnace to perform both fuel material recovery from damaged fuel rods by oxidative decladding and sinterability improvement of fuel powder by repetition of oxidative and reaction treatment. A horizontal type furnace provides only a diffusion-controlled reaction resulting in longer reaction time and decreasing amount of powder for oxidation and reduction, whereas a vertical type furnace with a submerged gas distributor gives rapid reaction due to flowing gas-solid contact by fluidization. For observation of fluidization behaviors of uranium oxides at room temperature, fluidized column was prepared with transparent cylindrical tube, pressure transmitter and gas flow meter. Number of size of orifice holes was determined by equations in Fluidization Engineering [D.Kunii, O. Levenspiel]. Before uranium oxide test, as surrogates, WO_2 (10.8 g/cm^3) and Ta_2O_5 (8.2 g/cm^3) powder similar to density of UO_2 (10.96 g/cm^3) and U_3O_8 (8.3 g/cm^3), respectively were used to achieve fluidization operation conditions in the region from minimum to expanded fluidization. Fluidization behaviors and pressure drop of powder bed was observed according to operation parameters such as gas velocity, number and size of orifice holes, and powder amount.

Acknowledgements

This study was supported by the National Research Foundation of Korea grant 2021M2A7A108 0748 funded by the Korean government.

Keywords: Fluidization Behavior, Pressure Drop, Uranium Oxides, Surrogates

Solvent Effect on Properties of Ag-Containing Aluminosilicate for Capturing Iodine

Jae Won Lee*, Ju Ho Lee, Chang Hwa Lee, and Yung Zun Cho

Korea Atomic Energy Research Institute (KAERI)

*njwlee@kaeri.re.kr

Ag-containing aluminosilicate sorbents capable of capturing iodine were prepared by sol-gelation from Na, Al, and Si alkoxides using co-solvent exchange, Ag/Na ion exchange, solvent exchange, and ambient-pressure drying. The $\text{Na}^+\text{AlSi-OH}$ gel was prepared using sodium methoxide (NaOMe): aluminum tri-sec-butoxide ($\text{Al}(\text{O}^i\text{Bu})_3$): tetraethyl orthosilicate (TEOS) molar ratios of 1.05:1:1, 1.3:1.1:1, 1.5:1.3:1. The solvent effect on textural properties such as Brunauer-Emmett-Teller (BET) surface areas and pore size distributions and Ag^0 particle sizes was investigated using water with high surface tension, isopropanol and n-heptane with low surface tension. The BET surface area, average pore size, and cumulative pore volume for sorbents strongly increased with decreasing surface tension of solvents and increasing Al/Si atomic ratios. In addition, Ag^0 particle sizes increased with decreasing surface tension of solvents.

Acknowledgements

This study was supported by the National Research Foundation of Korea grant 2021M2E3A1040059 funded by the Korean government.

Keywords: Solvent Effect, Silver, Aluminosilicate, Textural Properties

Improvement of the Evaluation Method for the Radioactivity in Spent Nuclear Fuel

Jongho Hong¹, Sangil Ahn^{1,*}, Jaechoon Lee¹, and Kiyoun Kim²

¹Korea Hydro & Nuclear Power (KHNP)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*ahn.sangil@khnp.co.kr

For the transport of spent nuclear fuel, it is necessary to evaluate the amount of radioactivity for each assembly and the total amount of radioactivity for each cask. Currently, KHNP is evaluating the radioactivity using the Express mode of the OrigenArp program in the SCALE6.1 code. Express mode is a method to evaluate the radioactivity assuming that it has been burned with the same power per cycle, and Detail mode is a method to evaluate the actual combustion history such as power and cooling time for each cycle.

For a total of 3,795 assemblies, including 1,391 assemblies for Kori Unit 1, 1,427 assemblies for Hanbit Unit 2, and 977 assemblies for Hanul Unit 3, the radioactivity was evaluated in Express mode and Detail mode, respectively, and the results were compared. As a result of the evaluation, it was confirmed that the results of the Express mode were evaluated more conservatively by 2.5~12.9% than that of the Detail mode. Accordingly, KHNP established a plan to change the evaluation method from Express mode to Detail mode in order to improve the accuracy of the radioactivity assessment results and eliminate conservatism.

Keywords: Spent Nuclear Fuel, Radioactivity

Discretization Uncertainty Estimation for Thermal Analysis of the Spent Fuel Transport Cask

Sung-yong Lee* and Dong-gyu Lee

KONES Corp.

*sylee@kones21.com

Thermal analysis and safety assessment of spent fuel transport cask are mainly conducted using commercial Computational Fluid Dynamics (CFD) codes based on Finite Volume Method (FVM). The reliability and predictability of CFD codes have greatly been improved by the development in the computer systems, and are widely used to calculate heat flow in complex structures that cannot be analyzed theoretically.

In the field of thermal analysis using the CFD code, it is important to clearly reflect the physical model of the transport cask, and a grid configuration suitable for the physical model is essential for accurate analysis. However, since there are no clear standard and guidelines for grid configuration and size, it is highly dependent on the user's insight. Spatial discretization errors result from the use of finite-width grids and the approximation of the differential terms in the model equations by difference operators. Since the user usually cannot change the truncation error order of a given discretization scheme, spatial discretization errors can only be influenced by the provision of optimal grids. Therefore, it is necessary to quantify the spatial discretization errors caused by the grid.

In the case of Orano TN's NUHOMS[®] MP197 transport cask, considering four grids for two sets, the temperature uncertainty of the neutron shield, which has the lowest margin at the limit temperature among transport cask components, was quantified by applying 5-step procedure of the Grid Convergence Index (GCI) method for the uncertainty estimation presented in ASME V&V 20-2009.

In the case of domestic spent nuclear fuel transport cask (KORAD21), neutron shield among the transport cask components has the lowest margin at the limited temperature. Accordingly, in this study, the temperature uncertainty of the neutron shield was quantified by applying GCI to three sets considering seven grids. As a result of the calculation, the uncertainty was less than $\pm 1^{\circ}\text{C}$, and the temperature of the neutron shield including the uncertainty was evaluated to be maintained below the limit temperature of 148°C .

Keywords: Spent Fuel, Transport Cask, Discretization Error, Grid Convergence Index, GCI

~~Spent Fuel Pool Criticality Evaluation by Applying Boron Credits~~

~~Yun Sik KIM[†], Dong Gyu Lee[†], and Ki Young Kim^{2,*}~~

~~[†]KONES Corp.~~

~~²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)~~

~~*kiyoungkim@khnp.co.kr~~

Withdrawal

The Quantification of Residual Water by Using the Drying Process Data in Dry Storage System

Geon-Hui Lee^{1,*}, Nam-Hui Lee¹, Kyung-Wook Shin¹, Gyung-Sun Chae¹, and Seung-Hwan Yu²

¹SAE-AN Engineering Corp.

²Korea Atomic Energy Research Institute (KAERI)

*ghui6@sae-an.co.kr

For Dry Storage of Spent Nuclear Fuel (SNF), all moisture must be removed from the dry storage canister through subjected to a drying process to ensure the long-term integrity. In NUREG-1536, the evacuation of most water contained within the canister is recommended a pressure of 0.4 kPa (3 torr) to be held in the canister for at least 30 minutes while isolated from active vacuum pumping as a measure of sufficient dryness in the canister. In the existing drying process, the determination of drying end point was determined using a dew point sensor indirectly. Various methods are being studied to quantify the moisture content remaining inside the canister. We presented a moisture quantification method using the drying process variables, like as temperature, pressure, and relative humidity operation data.

During the drying process, it exists in the form of a mixed gas of water vapor and air inside the canister. At this time, if the density of water vapor in the mixed gas discharged out of the canister by the vacuum pump is known, the mass of water removed by vacuum drying can be calculated. The canister is equipped with a pressure gauge, thermometer and dew point sensor. The density of water vapor is calculated using the pressure, temperature and relative humidity of the gas obtained from these sensors. First, calculate the saturated water vapor pressure, and then calculate the humidity ratio. The humidity ratio refers to the ratio of water vapor mass to the dry air mass. After calculating the density of dry gas, multiply the density by the humidity ratio to calculate the density of water vapor (kg/m^3). Multiply the water vapor density by the volume flow (m^3/s) to obtain the mass value of water (kg). The calculated mass value is the mass value obtained per second since it is calculated through the flow data obtained every second, and the amount of water removed can be obtained by summing all the mass values. By comparing this value with the initial moisture content, the amount of moisture remaining inside the canister can be estimated.

The validity of the calculations will be verified through an experimental test in the near future. We plan to conduct various research and development to quantify residual water, which is important to ensure the safety of the drying process for dry storage.

Acknowledgements

This work has supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (No. NRF-2021M2E1A1099409).

Keywords: Moisture Quantification, Residual Moisture, Dry Storage, Spent Nuclear Fuel

Criticality Impact Assessment of Fuel Rod Loss for the Development of Spent Fuel Acceptance Criteria

Janghwan Lim*, Seongki Lee, and Jongsung Yoo

KEPCO Nuclear Fuel (KEPCO NF)

*jhl@knfc.co.kr

Currently, as the saturation capacity of wet storage pool for spent nuclear fuel (SNF) of PWR in Korea has reached approximately 75%, Dry Storage Facilities (DSF) are necessary for sustainable operation of nuclear power plants. It is necessary to develop acceptance requirements for the delivery of SNF from reactor storage site to Centralized DSF. To do this end, the mechanical integrity of SNF is directly related to its repacking, retrieving, and transporting/handling performances. And also, this integrity is a key factor associated with the criticality safety that is connected to the damaged status of SNF.

According to the NUREG/CR-6835, the NRC expects that the potential for nuclear fuel failures will increase because of the increase of the fuel discharge burnup and the degradation of fuel and clad material properties. Due to such damages and/or degradation, the fuel rods in the fuel assembly may be extracted and empty for following treatments (transportation, storage, handling etc). This condition can have a detrimental effect on the criticality safety of SNF. Thus, this study investigated whether extracted and empty of damaged SNF rod affects criticality safety. In this analysis, it is assumed that up to four fuel rods are missed.

As a result of the analysis, As the number of fuel rods miss up to a certain number, the value of multiplication factor value of the fuel assembly increases. In addition, since the fuel rods located at the outermost layer contained relatively less fissile material than the fuel rods located center of the lattice, and neutrons were lost by the absorption material, the effective multiplication factor value gradually decreased. Nevertheless, the criticality safety was assessed to be maintained.

Acknowledgements

This work was supported by the Korea Spent Nuclear Fuel grant funded by the Korea government the Ministry of Trade, Industry & Energy (No. 2021040101002A).

Keywords: Spent Fuel, Criticality Safety Analysis, Missing Fuel Rod

Assessment of Potential Degradation Mechanisms to Resolve Integrity Issues of CANDU Spent Fuel

Kang Moon Lee*, Nam Ho Lee, Seong Ki Lee, and Jong Sung Yoo

KEPCO Nuclear Fuel (KEPCO NF)

*kmoonlee@knfc.co.kr

Maintaining fuel sheath integrity during dry storage is important. Intact sheath acts as the primary containment barrier for both fuel pellets and fission products over the dry storage periods and during subsequent fuel handling operations. In KNF, in-house fuel performance code was developed to predict the overall behavior of a fuel rod under normal operating conditions. It includes the analysis modules to predict temperature, pellet cracking and deformation, sheath stress and strain at the mid-plane of the pellet and pellet-pellet interfaces, fission gas release and internal gas pressure. The main focus of the code is to provide information on initial conditions prior to dry storage, such as fission gas inventory and its distribution within the fuel pellet, initial volumes of storage spaces and their locations, radial profile of heat generation within the pellet, etc.

To upgrade the developed code that address all the damage mechanisms, the first step was a review of the available technical information on phenomena relevant to fuel integrity. Potential degradation mechanisms that may affect sheath integrity of CANDU spent fuel during dry storage are: creep rupture under internal gas pressure, sheath oxidation in air environment, stress corrosion cracking (SCC), delayed hydride cracking (DHC), and sheath splitting due to UO_2 oxidation for a defective fuel. The failure by creep rupture, SCC or DHC is in the form of small cracks or punctures. The failure by sheath oxidation or sheath splitting due to UO_2 oxidation results in a gross sheath rupture.

The second step was to examine the technical bases of all modules of the in-house code, identify and extend the ranges of all modules to required operating ranges. This step assessed the degradation mechanisms for the fuel integrity. The objective of this assessment is to predict the probability of sheath through-wall failure by a degradation mechanisms as a function of the sheath temperature during dry storage.

Further improvements being considered include upgrades of the analysis module to achieve sufficient accuracy in key output parameters. The emphasis in the near future will be on validation of the in-house code according to a rigorous and formal methodology. The developed models provide a platform for research and industrial applications, including the design of fuel behavior experiments and prediction of safe operating margins for CANDU spent fuel.

Keywords: CANDU Spent Fuel, Fuel Performance Code, Potential Degradation Mechanism, Dry Storage

Ring Tensile Test Material Properties Calibration Using Optimization Technique

JaeYong Kim^{1,*}, HongRyoul Oh¹, and SungHoon Park²

¹Korea Atomic Energy Research Institute (KAERI)

²V-ENG

*kjdkjy@kaeri.re.kr

A tensile test is performed to obtain the mechanical property data of the spent fuel cladding. In general, the elastic modulus, elongation, yield stress, tensile stress, etc. are obtained by axial tensile test of cladding attaching an extensometer. However, due to the limitation in the number of specimens for spent nuclear fuel that can be made, the ring tensile test (RTT) whose required length of the specimen is short is mainly performed.

In the case of RTT, an extensometer or strain gauge cannot be attached because the gauge part of the specimen is formed around the cladding and is short. In addition, since a load is applied in the radial direction of the cladding, a curved portion of the circular cladding is spread out and becomes straight, and then the cladding is tensioned. For this reason, it is difficult to obtain the stress-strain curve directly from the RTT results.

Isight, which is used to identify the optimization design parameters, was used to build an optimization process that minimizes the difference between the RTT and the analysis to estimate the material property. For this, the elastic modulus, plastic strain, and the radius of the RTT jig were taken as fixed variables. As variables, isotropic hardening data and plastic stress were taken. The objective function was taken as the minimization of the area difference of the load-displacement curve obtained from the tests and analysis, of the difference in the magnitude of the maximum reaction force, and of the difference in the location where the maximum reaction force occurred.

Optimization workflow was configured in the following order. First, using the calculator component, plastic stress design variables were created. Next, ABAQUS was placed to perform analysis using design variables, and the reaction force or displacement was calculated. After that, the reaction force was calculated considering the 1/4 symmetry condition using the script component. After that, the data matching component performed quantitative comparison of test and analysis data. Finally, by utilizing the exploration component, the plastic stress design variable that minimizes the difference in the objective function was obtained by automatically changing six optimization algorithms.

In this paper, the constructed optimization process and the obtained plastic stress by applying it to the SUS316 RTT results are briefly described. The established optimization process can be utilized to obtain mechanical property from the results of the cladding RTT of spent nuclear fuel or new material.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (20217810100050, Development of Safety Enhanced Nuclear Core Technology for APR).

Keywords: Ring Tensile Test, Fuel Cladding, Mechanical Property Calibration, Optimization Technique, Isight

Review on Non-Destructive Burnup Evaluation Method for Spent Nuclear Fuel

Daegwon Eun*, Jae-Hun Ko, Moonoh Kim, and Dong-Gyu Lee

KONES Corp.

*mephoenix@kones21.com

In ROK, when designing a spent nuclear fuel (SNF) storage facility and cask, criticality safety analysis is performed assuming that the SNF is a fresh fuel in order to ensure conservatism. Storage and transportation capacity can be increased by more than 30% by applying the burnup credit, but it has not been applied to the management of SNF. On the other hand, currently in criticality safety analysis, average burnup value is applied to axial burnup profiles, and it is not conservative because burnup of the middle of SNF is greater than average value. Thus, measuring burnup of SNF with high accuracy contributes to the economics and safety of the management of SNF. In this paper, non-destructive burnup evaluation methods for SNF are reviewed in order to study how to measure burnup more accurately.

Gamma ray spectrometry and neutron counting have been used as non-destructive burnup evaluation methods of SNF. Gamma spectrum analysis uses the ratio of Cs-134/Cs-137 or Eu-154/Cs-137. The ratio of Cs-134/Cs-137 is used to SNF with cooling time less than 20 years, and the ratio of Eu-154/Cs-137 is used to SNF with cooling time more than 20 years due to their half-life. In spectrum analysis, detector sensors with high efficiency and energy resolution are needed to clarify each spectrum. High-purity germanium (HPGe) detector has high energy resolution. However, it is not suitable for the analysis of the SNF in the spent fuel pool because it requires separate cooling system and large volume. Thus, CdZnTe (CZT) detector, which has medium energy resolution, is used as a detector of gamma ray spectrometry for the analysis of the SNF in the spent fuel pool. Recently, LaBr₃ detector has been commercialized. Although it is difficult to compare clearly due to different conditions such as detector volume and crystal size, LaBr₃ detector showed better resolution than CZT in the entire energy region.

Neutron counting method has a large error compared to gamma spectrometry because the neutron flux is lower than gamma ray, and neutron absorption reaction, induced fission, and pool environment have to be considered. Large quantity of gamma energy is deposited in the detector by the fission fragments near the SNF. Therefore, fission chambers, which have the highest insensitivity to gamma rays, must be used as neutron detector in order to avoid noise from gamma rays.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. 20222B1010060).

Keywords: Burnup Evaluation, Non-destructive, Gamma Spectrometry, Neutron Counting, Spent Nuclear Fuel

Applicability Evaluation of Ramberg-Osgood Equation as New Cladding Plastic Material Model

JaeYong Kim^{1,*}, YongSik Yang¹, and JuYeop Park²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Institute of Nuclear Safety (KINS)

*kjkjy@kaeri.re.kr

For the spent fuel modeling, the plastic model of the cladding used in FRAPCON uses the $\tilde{\sigma} = K\tilde{\epsilon}^n \left[\frac{\dot{\epsilon}}{10^{-3}} \right]^m$ format. Strength coefficient (K), strain hardening exponent (n), strain rate sensitivity constant (m) are derived as the function of temperature. The coefficient m related to the strain rate shows dependence on the strain rate only in the α - β phase transition section, 1,172.5~1,255 K. But this is the analysis range of the FRAPTRAN code, which is an accident condition nuclear fuel behavior evaluation code. It does not apply to evaluate spent fuel. This coefficient in FRAPCON is used as a constant value (0.015) below 750 K (476.85°C), and at a temperature above 750 K, it is assumed that it is linearly proportional to the temperature without considering the strain rate dependence, also. In order to confirm the effect of strain rate, multiple test data performed under various conditions are required. However, since the strain rate dependence is not critical and test specimen limitation in the case of spent fuel, it is needed to replace with a new plastic model that does not include the strain rate term.

In the new plastic model, the basic form of the Ramberg-Osgood equation (RO equation) is the same as $\tilde{\epsilon} = \frac{\tilde{\sigma}}{E} + K_r \left(\frac{\tilde{\sigma}}{E} \right)^{n_r}$. If the new variable α is defined as $\alpha = K_r (\sigma_o/E)^{n_r-1}$, this equation can be transformed into $\tilde{\epsilon} = \frac{\tilde{\sigma}}{E} + \alpha \frac{\tilde{\sigma}}{E} \left(\frac{\tilde{\sigma}}{\sigma_o} \right)^{n_r-1}$.

The procedure for expressing the stress-strain curve of the cladding with the RO equation is as follows. First, convert the engineering stress-strain into true stress-strain. Second, using a data analysis program such as EXCEL or ORIGIN, obtain the slope of the linear trend-line on the linear part and use it as the elastic modulus. Third, using the 0.2% offset method, find the yield point and the yield stress. Finally, using the solver function of EXCEL, find the optimal values of α and n_r that minimize the sum of errors.

The applicability of the suggested RO equation was evaluated using the results of the Zircaloy-4 plate room temperature tensile test performed by the KAERI and the Zircaloy cladding uniaxial tensile test results presented in the PNNL report. Through this, the RO equation was able to express the tensile test results within the uncertainty range of ± 0.005 .

In this paper, the RO equation is suggested as a new plastic model with limited test data due to the test specimen limitation of spent fuel and its applicability is confirmed.

Acknowledgements

This work has been carried out supported by the Korea Foundation of Nuclear Safety R&D Program (2106002-0222- WT111).

Keywords: Cladding Plastic Material Model, Ramberg-Osgood Equation, FRAPCON, Tensile Test, Stress-Strain Curve

Estimation Model of Residual Water Evaporation During Vacuum Drying

Seunghwan Yu* and Kyoung-sik Bang

Korea Atomic Energy Research Institute (KAERI)

*shyu80@kaeri.re.kr

For the decommissioning or continuous long-term power generation of nuclear power plants, it is necessary to transfer the spent nuclear fuel from the wet storage pool to the dry storage. Spent nuclear fuel should go through the drying process, which is the first step of dry storage. The most important part in the drying process is the removal of the residual water. The spent fuel might be stored in a dry storage system for a long time. The integrity of internal components and spent fuel cladding should be maintained during the storage period. If residual water is present, problems such as aging of metal materials, oxidation of cladding, and the hydride-reorientation could occur.

The presence or absence of residual water after vacuum drying is evaluated by pressure. If there is residual water in the vacuum drying process, it evaporates easily at low pressure to form water vapor pressure and the internal pressure rises. In the recent EPRI High burn up demonstration test, the gas inside the canister that satisfied the dryness criteria was extracted and analyzed. It showed that the water content was higher than the expected value. We are conducting verification studies on the pressure evaluation method, which is an indirect evaluation method of vacuum drying.

The vacuum drying test was performed on small specimens at Sandia National Laboratory, and quantitative residual water evaluation was also performed. The report did not mention a detailed method for the assessment of residual water. Based on the test results of SNL, direct residual water evaluation was performed using energy balance. If the dryness criteria were satisfied, the quantitative amount of residual water was also evaluated. As a result, almost the same result as the evaluation result of SNL was derived, and it was confirmed that most of the water was removed when the dryness criteria was satisfied.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085226).

Keywords: Vacuum Drying, Spent Nuclear Fuel, Quantitative Evaluation, Residual Water, Energy Balance

Development of Damaged Fuel Canister for Dry Storage

Seo-Jeong Lee*, Kwang-Ho Lee, and Eun-Ki Lee

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*s.jeong2@khnp.co.kr

Due to the saturation of the on-site storage capacity of spent nuclear fuel within a few years, dry storage facility should be introduced. However, it is unclear when to start operating the dry storage facility, so in case of Kori Unit 1, which is being decommissioning, the spent fuel must be stored in the spent fuel pool of another power plant. In addition, in the case of damaged fuel, it is impossible to transfer and store it with general handling methods. Therefore, a damaged fuel canister (DFC) should be able to handle damaged or failed fuel as intact fuel, and both wet and dry storage should be possible.

The canister developed by Korea Hydro & Nuclear Power is designed to satisfy criticality, shielding, cooling performance, and structural integrity in accordance with NUREG-1536 and 2215. In addition, it can be handled as existing fuel handling devices rather than new handling tools.

Fastening of the DFC lid and body in the spent fuel pool is possible with a hexagonal socket wrench, one of the fuel repair tools. And it is designed to facilitate visual identification of whether it is fastened or not. The lifting method for transferring DFC to another facility is the same as the nuclear fuel lifting method. And a unique sealing and mesh structure of the lid and body is devised to completely block leakage of nuclear fuel fragments of 0.2 mm or more during vacuum drying for dry storage.

The usability of DFC has been verified through test operation of the prototype, and it will be manufactured before discharging spent fuel for the decommissioning of Kori Unit 1.

Keywords: DFC, Dry Storage, Decommissioning, Fuel Handling, Fastening, Sealing

Quantitative Analysis of Sr and Cs in LiCl-KCl Molten Salt by Laser-Induced Breakdown Spectroscopy (LIBS)

Hyeongbin Kim¹, Yunu Lee², Wonseok Yang^{1,2}, and Sungyeol Choi^{1,*}

¹Seoul National University (SNU)

²Korea Advanced Institute of Science and Technology (KAIST)

*choisys7@snu.ac.kr

During electrorefining, fission products, such as Sr and Cs, accumulate in a eutectic LiCl-KCl molten salt and degrade the efficiency of the separation process by generating high heat and decreasing uranium capture. Thus, the removal of the fission products from the molten salt bath is essential for reusing the bath, thereby reducing the additional nuclear waste. While many studies focus on techniques for selective separation of fission products, there are few studies on processing monitoring of those techniques. In-situ monitoring can be used to evaluate separation techniques and determine the integrity of the bath. In this study, laser-induced breakdown spectroscopy (LIBS) was selected as the monitoring technique to measure concentrations of Sr and Cs in 550°C LiCl-KCl molten salt. A laser spectroscopic setup for analyzing high-temperature molten salts in an inert atmosphere was established by coupling an optical path with a glove box. An air blower was installed between the sample and lenses to avoid liquid splashes on surrounding optical products caused by laser-liquid interaction. Before LIBS measurements, experimental parameters such as laser pulse energy, delay time, and gate width were optimized for each element to get the highest signal-to-noise ratio of characteristic elemental peaks. LIBS spectra were recorded with the optimized conditions from LiCl-KCl samples, including individual elements in a wide concentration range. Then, the limit of detections (LODs) for Sr and Cs were calculated using calibration curves, which have high linearity with low errors. In addition to the univariate analysis, partial least-squares regression (PLSR) was employed on the data plots to obtain calibration models for better quantitative analysis. The developed models show high performances with the regression coefficient R^2 close to one and root-mean-square error close to zero. After the individual element analysis, the same process was performed on samples where Sr and Cs were dissolved in molten salt simultaneously. The results also show low-ppm LODs and an excellent fitted regression model. This study illustrates the feasibility of applying LIBS to process monitoring in pyroprocessing to minimize nuclear waste. Furthermore, this high-sensitive spectroscopic system is expected to be used for coolant monitoring in advanced reactors such as molten salt reactors.

Acknowledgements

This work was supported through the National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT directly (grant number: NRF-2022M2D4A1052797).

Keywords: Pyroprocessing, Laser-induced Breakdown Spectroscopy, In-situ Monitoring, Multi-element analysis, Partial Least-squares Regression

Seismic Analysis of Concrete Pad in the Dry Storage Facilities According to Soil Properties

Seungpil Kim and Sang Soon Cho*

Korea Atomic Energy Research Institute (KAERI)

*sscho96@kaeri.re.kr

Some Spent Fuel Pools (SFPs) will be full of Spent Nuclear Fuels (SNFs) within several years. Because of this reason, building interim storage facilities or permanent disposal facilities should be considered. These storage facilities are divided into wet storage facilities and dry storage facilities. Wet storage facility is a method of storing SNF in SFP to cool decay heat and shielding radiation, and dry storage facility is a method of storing SNF in a cask and placing on the ground or storage building. However, wet storage facilities have disadvantages in that operating costs are higher than that of dry storage facilities, and additional capacity expansion is difficult. Dry storage facilities have relatively low operating costs and are relatively easy to increase capacity when additional SNFs need to be stored. For this reason, since the 1990s, the number of cases of applying dry storage facilities has been increasing even abroad.

Dry storage facilities are divided into indoor storage facilities and outdoor storage facilities, and outdoor storage facilities are mostly used to take advantage of dry storage facilities. In the case of outdoor storage facilities, the cask in which SNFs are stored is placed on a designed concrete pad. During this storage, the boring heat generated by SNFs cools into natural convection and the cask shields the radiation that SNFs generates. However, if an accident such as an earthquake occurs and the cask overturns during storage, there may be a risk of radiation leakage. Such a tip-over accident may be caused by the cask slipping due to the vibration of an earthquake, or by not supporting the cask properly due to a problem in the concrete pad. Therefore, in the case of outdoor dry storage facilities, it is necessary to evaluate the seismic safety of concrete pads.

In this paper, various soil conditions were applied in the seismic analysis. Soil conditions were classified according to the shear wave velocity, and the shear wave velocity was classified according to the ground classification criteria according to the general seismic design (KDS 17 10 00). The concrete pad was designed with a size that 8 casks can be arranged at regular intervals, and 11# reinforcing bars were used for the design of the internal reinforcement of the concrete pad according to literature research. The cask was designed as a rigid body to shorten the analysis time. The soil to which the elastic model was applied was designed under the concrete pad, and infinite elements were applied to the sides and bottom of the soil. The effect on the concrete pad and cask by applying a seismic wave conforming to RG 1.60 to the bottom of the soil was analyzed with a finite element model.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085229).

Keywords: Concrete Pad, Seismic Analysis, Finite Element Analysis, Earthquake Accident, Soil Design

DeCART2D/MASTER Core Follow Calculation for Hanbit Unit 3 and Comparison With McCARD Single Fuel Assembly Burnup Analyses

Jeong Woo Park, Seung-Ah Yang, and Ho Jin Park*

Kyung Hee University

*parkhj@khu.ac.kr

In the design of a spent-fuel (SF) storage, the consideration of burnup credit brings the benefits in safety and economic views. According to it, various SF burnup measurement systems have been developed to estimate high fidelity burnup credit, such as FORK and SMOPY. Recently, there are a few attempts to localize the SF burnup measurement system in South Korea. For the localization of SF burnup measurement systems, it is very important to build the isotope inventory data base (DB) of various kinds of SFs.

In this study, we performed DeCART2D/MASTER core follow calculations and McCARD single fuel assembly (FA) burnup analyses for Hanbit unit 3 and confirmed the characteristic of the isotope inventory over burnup. Firstly, the core follow calculations for Cycles 1~7 were performed using DeCART2D/MASTER code system. The core follow calculation is very realistic and practical because it considers the design conditions from its nuclear design report (NDR). Secondly, the Monte Carlo burnup analyses for single FAs were conducted by the McCARD Monte Carlo (MC) transport code. The McCARD code can utilize continuous energy cross section library and treat complex geometric information for particle transport simulation. Accordingly, the McCARD code can provide accurate solutions for burnup analyses without approximations, but it needs huge computing resources and time burden to perform whole-core follow calculations.

Therefore, we will confirm the effectiveness of the single McCARD FA burnup analyses by comparing the DeCART2D/MASTER core follow results with the McCARD solution. From the results, the use of single FA burnup analyses for the establishment of the DBs will be justified. Various FAs, that have different ^{235}U enrichments and loading pattern of fuel rods and burnable absorbers, were considered for the burnup analyses. In addition, the results of the sensitivity analyses for power density, initial enrichment, and cooling time will be presented.

Acknowledgements

This work was partly supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the korea government (MOTIE) (20222B10100060, Development of On-site Burn-up Detection System for the Spent Fuel).

Keywords: Spent Fuel Assembly Nuclide Number Density, Burnup Analysis, Core Follow Calculation, DeCART2D/MASTER, Single FA Burnup Calculation, McCARD

Thermal Performance Sensitivity Analysis for Modeling of Spent Fuel Assembly in Transport System

Kun-ho Chun*, Joung-yeul Lee, and Seong-ki Lee

KEPCO Nuclear Fuel (KEPCO NF)

*khchun@knfc.co.kr

This study is to investigate fuel cladding temperature in a transport system for the purpose of developing a methodology for evaluating the thermal performance of spent fuel. Detailed temperature analysis in the transport system is important because the degradation mechanism of the fuel cladding is generally sensitive to temperature and temperature history. In such a system, the magnitude of the temperature change is determined by examining the temperature sensitivity of fuel assemblies and system components including fuel cladding temperature, considering the material properties, component specifications, component aging mechanism, and heat transfer mechanism. The sensitivity analysis is performed using heat transfer models by computational fluid dynamics for the horizontal transport system. The heat transfer within the system by convection, conduction and thermal radiation is calculated by thermal-hydraulic analysis code FLUENT. The calculation region is divided into a basket cell and a transport cask. The thermal analysis of the basket cell is for predicting the fuel cladding temperature. And the reason for analyzing the transport cask is to provide the boundary condition for the basket cell by reflecting the external environmental conditions. Here, the basket cell containing the spent fuel assembly is modeled on the homogeneous effective thermal conductivity.

The purpose of this analysis is to evaluate fuel cladding temperatures for the following four main items. That is the effect of surface emissivity changes in basket due to the oxide layer of the fuel cladding, the effect of degradation of the canister backfill helium gas, the effect of fuel assembly position in basket cell on fuel cladding and basket temperatures in canister, and the effect of using the homogeneous effective thermal conductivity model instead of the fuel assembly in basket cell.

As a result of the analysis, the maximum temperatures in basket cells are evaluated for the above four items. Thermal margins for each item are investigated for thermal performance requirements (e.g., peak clad temperature below 400°C).

Keywords: Spent Fuel Assembly, Spent Fuel Transport System, Thermal Performance, Sensitivity Analysis, Heat Transfer

A Study on the Design Parameters of the Off-gas Trapping System

Byungsuk Park*, Seokmin Hong, Juho Lee, Jaewon Lee, and Changhwa Lee

Korea Atomic Energy Research Institute (KAERI)

*nbspark@kaeri.re.kr

The Korea Atomic Energy Research Institute is developing a nuclide management process that separates high heat, high mobility, and long half-life nuclides that burden the disposal of spent fuel, and disposes of spent fuel by nuclide according to the characteristics of each nuclide. Various off-gases (volatile and semi-volatile nuclides) generated in this process must be discharged to the atmosphere below the emission standard, so an off-gas trapping system is required. In this study, we introduce the analysis results of the parameters that affect the design of the off-gas trapping system. The analyzed contents are as follows. The physical quantities of the Cs, Tc/se, and I trapping filters according to the amount of spent nuclear fuel, the maximum exothermic temperature of the Cs trapping filter and the absorbed dose by distance by Cs radioactivity were analyzed according to the amount of spent nuclear fuel. In addition, a three-dimensional CFD (Computational Fluid Dynamics) analysis was performed according to operating parameters by simply modeling the off-gas trapping system, which is easy to modify mechanical design parameters. It is considered that the analysis results will greatly contribute to the development of the off-gas trapping system design requirements.

Acknowledgements

This work was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT (NRF-2021M2E3A1040059).

Keywords: Nuclide, Off-gas, Trapping, Design, Parameter, CFD

Lessons Learned in Improvement Work of Fuel Handling Equipment at Operating Nuclear Power Plants

Sang-Gyoon Chang^{1,*}, Won-Bae Park¹, Min-Gyu Kim¹, and Dae-Hyuk Kwon²

¹KEPCO Engineering & Construction Company (KEPCO E&C)

²Korea Hydro & Nuclear Power (KHNP)

*sgchang@kepc-eec.com

The purpose of this study is to provide lessons learned in the experience of improvement work of fuel handling equipment at operating nuclear power plants. The upgrade of fuel handling equipment for safety enhancement and performance improvement has been going on for 15 years since the early 2000's. The main goal is to increase fuel loading/unloading capability of the equipment from about 2.5 fuel assemblies per hour to more than six (6). It is achieved with sequential operations of three (3) fuel handling equipment, which consists of the refueling machine, the fuel transfer system and the spent fuel handling machine.

The scope of the upgrade for fuel handling equipment is summarized as follows. The PC data control system based on PLC for interlocks and high speed motor drive system is introduced for better operating efficiency. The motors and drives for bridge, trolley, and hoist are replaced with AC servomotors and drivers, respectively. The fuel transfer system has an auto-initiation feature operating from the refueling machine or the spent fuel handling machine. The winch motor and drive for the carriage of fuel transfer system is also replaced with AC servomotors and drivers. And some of HPU (hydraulic power units) equipment for each building (reactor containment building and fuel handling building) are replaced to improve their function.

The considerations for improvement of fuel handling equipment are as follows. 1) Fuel handling should be consistent with the instructions provided by the fuel designer and/or manufacturer, which are for Standard type fuel and Westinghouse type fuel, used in domestic nuclear power plants. Each fuel has unique design characteristics, which are PLC setpoints for overload and underload, slow speed zones for a bridge, trolley and hoist, allowable acceleration/deceleration value in handling, hoist elevation and manual speed in off-index operation at reactor. 2) The interlock system should be designed in accordance with design criteria specified by the utilities of nuclear power plant. 3) Performance should be improved according to the operating characteristics of the fuel handling equipment. High-speed and optimization of FTS upender and carriage are essential for operating performance so that its modification should be considered first. And the low speed and range in the operation mechanism of the hoist should be designed to comply with guidelines. 4) The accident analysis through self-diagnosis function and operation history in modification at domestic operating nuclear power plants would be good lessons learned. It is advisable to utilize such various information as it can help to improve reliability of nuclear fuel handling operation and power plant operation rate.

Keywords: Fuel Handling Equipment, Modification and Upgrade, Fuel Transfer Capability, Refueling Equipment

A Preliminary Consideration to Categorize Classes of Major SSCs of a Dry Storage System for SNF Based on Importance to Safety

Sang Hyeok Lee and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

As the amount of on-site Spent Nuclear Fuel (SNF) in storage increases due to the continued operation of Nuclear Power Plants (NPPs) in Korea, the on-site wet storage pool is expected to become saturated. Therefore, a facility for safely storing the spent nuclear fuel is required so that there is no problem with operation of the NPP until permanent disposal of SNF. Prior to the construction of such a facility, the safety analysis of the interim storage facility and verification of the safety of the spent fuel storage system (e.g. cask, silo) to be used are required according to Article 63 of the Nuclear Safety Act. In this process, analysis of the Structures, Systems, and Components (SSCs) of the storage system is needed. Based on the analysis, it is necessary to efficiently classify SSCs that are important to safety in order to differentiate management that more thoroughly manages those important to safety.

In Korea, according to the notice of the Nuclear Safety and Security Commission, the components performing essential safety functions for the safe storage of spent fuel storage system are to be classified as “important safety equipment”. 10 CFR Part 72, a federal regulation related to interim storage facilities in the United States, also requires the identification of SSCs that fall under “Important to Safety (ITS)”, which is like domestic case. In addition, it has been confirmed that there are cases in which detailed classification according to Reg Guide 7.10 and NUREG-CR/6407 is added in Safety Analysis Report. However, these existing classification methods are not only classified as a single grade except for the method according to the Reg guide, but all are classified according to a qualitative standard. Qualitative criteria may cause ambiguity in judgment, resulting in subjective judgment of the person who proceeds in the classification process.

Therefore, in this study, a new classification method is proposed to solve the problem according to the qualitative classification method. Assessing the level of radiological harm to the general public due to the assumption of failure of SSC in the spent fuel storage system is used as a quantitative evaluation standard.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of the Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2021040101002B20220133).

Keywords: Spent Nuclear Fuel Cask, Classification, Graded Approach, Radiological Harm

Effect of Sat Stirring on Phosphorylation of Uranium and Rare Earths in Molten Salt

Joon-Bo Shim*, Hansoo Lee, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*njbshim@kaeri.re.kr

Under the circumstance of energy transition policy of the previous government in which nuclear energy portion will be gradually reduced, some R&D study looking for alternatives other than Pyro-SFR recycling could be very valuable and timely suitable. New alternative study started to evaluate the possibility of it if there are some advantages in terms of waste burden in case that the spent fuel are appropriately treated and disposed of in a disposal site, instead of recycling of spent nuclear fuels (SNF). The alternative study separate the fission products (minor actinides and rare earths) from SNF in a molten salt medium. The molten salt coming from the alternative study is radioactive and heat generating because it contains the fission products chlorides. It is necessary to collect the fission products from the waste molten salt for minimization of the high-level waste volume and to generate a final waste form containing the fission products compatible to the disposal site. Based on the results of a review for various precipitation methods, phosphorylation (phosphate precipitation) of metal chlorides selected as a proper treatment method for recovering of the fission products in a molten salt. Phosphate precipitation has the potential for removing most of fission product elements from a molten salt arising from the treatment of spent nuclear fuel. The performance of phosphate precipitation method evaluated using a salt mixture with the actinide and rare earth chlorides. The molten salt containing uranium as surrogate of the actinides and three rare earths (Nd, Ce, La) chloride was used for testing a phosphate precipitation method at experimental condition (temperature 500°C, salt stirring 200~300 rpm, and 1~1.2 eq. of phosphorylation agent). A cyclic voltammetry (CV) method monitored in-situ phosphate precipitation progress for determining the precipitation rate and conversion ratio evaluated. The phosphorylation reaction increased greatly at a salt stirring 300 rpm.

Keywords: Uranium, Rare Earths, Phosphate Precipitation, Molten Salt, Spent Nuclear Fuel

Experiment for a Collection Method of Phosphorylation Products From Molten Salt

Joon-Bo Shim*, Ju Ho Lee, Sang-Woon Kwon, and Chang Hwa Lee

Korea Atomic Energy Research Institute (KAERI)

*njbshim@kaeri.re.kr

Based on the results of a review for various precipitation methods phosphorylation (phosphate precipitation) of metal chlorides considered as a proper treatment method for recovering of the fission products in a molten salt. In previous precipitation tests, the powder of lithium phosphate (Li_3PO_4) added into LiCl-KCl molten salt containing metal chlorides as a precipitation agent. The reaction of metal chlorides containing actinides and rare earths to recover with lithium phosphate in a molten salt known as solid-liquid reaction. The powder of lithium phosphate disperse in a molten salt by stirring thoroughly in order to enhance the precipitation reaction. As a result, metal phosphates as the reaction products precipitate on the bottom of the vessel and cutting at the lower part of the salt ingot considered as one of the recovery method of the precipitates. Recently, the vacuum distillation of upper part of the salt proposed as another recovering method. Cutting method of precipitate at the lower part of the salt ingot would be difficult to handle the increased size of the salt ingot produced from the practical scale equipment. In this presentation, a new method for collecting the precipitates of phosphorylation reaction into a small vessel is introduced with test results in a molten salt containing uranium and rare earths such as Nd, Ce, and La. As the first step of a series of test lithium phosphate ingot was prepared by melting the powder at a temperature $1,300^\circ\text{C}$, and the ingot put into LiCl-KCl molten salt at 500°C for more than three hours to examine the shape of ingot to be deformed or not. The phosphorylation experiments using lithium phosphate ingots carried out to collect the metal phosphate precipitates and the test result of this new method was feasible.

Keywords: Uranium, Rare Earths, Phosphate Precipitation, Molten Salt, Spent Nuclear Fuel

Development of Code for Pressure Resistance Evaluation of Spent Nuclear Fuel Transport Cask Submerged in the Sea

Jiyoung Park, Hakyong Jang, and Sanghoon Lee*

Keimyong University

*shlee1222@gmail.com

In case a spent nuclear fuel transport cask is lost in the sea due to an accident during maritime transport, it is necessary to evaluate the critical depth by which the pressure resistance of the cask is maintained. A licensed type B package should maintain the integrity of containment boundary under water up to 200 m of depth. However, if the cask is damaged during accidents of severity exceeding those of design basis accidents, or it is submerged in a sea deeper than 200 m, detailed analyses should be performed to evaluate the condition of the cask and possible scenarios for the release of radioactive contents contained in the cask. In this work, models to evaluate pressure resistance of an undamaged cask in the deep sea are developed and coded into a computer module. To ensure the reliability of the models and to maintain enough flexibility to account for a variety of input conditions, models in three different fidelities are utilized. A very sophisticated finite element analysis model is constructed to provide accurate response of containment boundary against external pressure. A simplified finite element model which can be easily generated with parameters derived from the dimensions and material properties of the cask. Lastly, mathematical formulas based on the shell theory are utilized to evaluate the stress and strain of cask body, lid and the bolts. The models in mathematical formula will be coded into computer model once they show good agreement with the other two model with much higher fidelity. The evaluation of the cask was largely divided into the lid, body, and bottom, bolts of the cask. It was confirmed that the internal stress of the cask was increased in accordance with the hydrostatic pressure. In particular, the lid and bottom have a circular plate shape and showed a similar deformation pattern with deflection at the center. The maximum stress occurred where the lid was in the center and the bottom was in contact with the body. Because the body was simplified and evaluated as a cylinder, only simple compression without torsion and bending was observed. The maximum stress occurred in the tangential direction from the inner side of the cylinder. The bolt connecting the lid and the body was subjected to both bending and tension at the same time, and the maximum stress was evaluated considering both tension and bending loads. In general, the results calculated by the formulas were evaluated to have higher maximum stresses than the analysis results of the simplified model. The results of the maximum stress evaluation in this study confirms that the mathematical models provide conservative results than the finite element models and can be used in the computer module.

Keywords: Spent Nuclear Fuel, Transport Cask, Maritime Transport, Hydrostatic Pressure, Maximum Stress, Simplified Cask Model

A GoldSim Model for Evaluating the Effectiveness of Radionuclide Management Process on Environmental Burden Reduction

Hong Jang*, Hun Suk Im, Hyo On Nam, Won Il Ko, and Yong Zun Cho

Korea Atomic Energy Research Institute (KAERI)

*janghong@kaeri.re.kr

Considering the domestic condition with small land area and high population density, it is necessary to develop technology that can reduce the disposal area than the deep geological disposal method. For this, KAERI is developing a nuclide management process that can reduce the environmental burden of spent fuel, and establishing an evaluation model that can evaluate the performance of various process options. It is expected that an optimal option of the nuclide management process can be derived from disposal perspective by applying the evaluation model.

The mass flow between processing steps of the radionuclide management process is the basic quantity required to quantify the evaluation criteria. Therefore, we built a generalized block model on GoldSim, which can simulate mass flow of various radionuclide management process options. In addition to the mass flow, this model was established to derive the amount of wastes generated by each processing step, the composition of nuclides, and radiological properties (decay heat, radioactivity, etc.). The mass flow and waste property derived from the models are closely related to the factors that determine the area of disposal concepts. Based on this, a disposal area calculation model was established as a model to evaluate the effectiveness of the radionuclide management process on environmental burden reduction.

For verification, three process options, which can manage radionuclides having high decay heat (Cs, Sr) or large volume (U), were selected and evaluated as reference processes. And two disposal options, deep geological disposal and deep borehole disposal concepts were considered to be linked with the processes. As a result, it was confirmed that the disposal area could be reduced in the process separating radionuclides having high decay heat. In the future, other evaluation models for economic viability and safety will be added in the GoldSim model.

Keywords: Radionuclide Management Process, Performance Evaluation Model, Goldsim Model

Strategic Study on Various Optional Treatment Processes for Disposal of Spent Nuclear Fuel

Hun-Suk Im*, Hong Jang, Hyo-On Nam, Won-Il Ko, and Yong Zun Cho

Korea Atomic Energy Research Institute (KAERI)

*ihs95@kaeri.re.kr

As an alternative technology for the efficient disposal of spent nuclear fuel, various process flows can be selected based on the recovered and separated radioactive nuclide group. This is to examine the efficiency of the disposal area of spent nuclear fuel when various disposal technologies and several treatment processes are applied to spent nuclear fuel, compared to the deep geological disposal of burying the entire spent fuel in the ground. Above all, the biggest advantage of the optional treatment processes is that it can be applied to various disposal methods (deep borehole disposal, deep geological disposal) because it can process spent fuel in various sizes and separate into some groups according to the properties of radionuclides. These optional processes are not new technology and currently available as of today, and the level is classified based on the stepwise separation of high heat emission nuclides and long half-life nuclides. This is to increase the efficiency of the disposal of spent nuclear fuel by separating and managing high-risk radionuclides separately. Relatively various optional processes are possible depending on the level, and characteristic analysis is performed on wastes treated with alternative technologies. The mass balance for each option process is completed, and the amount of waste is also calculated accordingly. These are used as basic data for waste disposal area and economic evaluation. Besides it is easy to process spent fuel of various sizes suitable for deep geological disposal or deep borehole disposal technology when an optional treatment technology is applied to spent fuel. However, since this selective process is based on the process structure constructed in a broad framework, it is considered that additional follow-up studies are needed not only on detailed technology but also on the flow and amount of waste.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E3A1040058).

Keywords: Nuclear Spent Fuel, Deep Geological Disposal, Pre-treatment, Option Process

Preliminary Assessment of Radiation Effects of Spent Fuel Transport to Interim Storage

Yunhui Lee, Chang-Lak Kim*, and Sanghwa Shin

Kepeco International Nuclear Graduate School (KINGS)

*clkim@kings.ac.kr

The saturation rates of the spent fuel (SF) wet storage at the Kori Nuclear Power Plant (NPP), Hanbit, and Hanul are 83.3%, 74.2%, and 80.8% as of the fourth quarter of 2021. The storages of Kori NPP and Hanbit NPP are expected to be saturated in 2031, and Hanul is expected to be saturated in 2032. Therefore, the construction of an interim storage facility to store the SF temporarily stored in the NPP was planned, and preparations for the safe transport of the SF are required. In this paper, radiological preliminary assessment using NRC-RADTRAN in the process of sea transport of SF from the wet storage or ISFSI of the Hanbit NPP to the optional interim storage facility was performed.

Since domestic SF transport vessels are not currently in operation, the specifications of the UK Pacific Grebe vessel which can carry up to 20 casks were used. The transport cask used the specifications of KORAD-21, a transport container developed in Korea. Because it can carry more SF assemblies than the existing KN-18. In addition, a land transport safety test was conducted in 2020 and a sea transport test is planned.

The sea transport route was entered by referring to the transport route of domestic low and intermediate level waste. The accidents rate was calculated using statistics on maritime accidents from 2017 to 2021. The probability accidents along the transportation route were evaluated as $3.152E-10$.

When transporting to an interim storage facility, the SF expected to be the main transport target was selected as WH 17X17, combustion 45,000 MWD/MTU, and concentration of 4.5%. The source term was calculated and entered according to this data and the release fraction was entered with reference to the DOE report.

In the case of normal transport without accident, the individual dose of the crew member and public residents were estimated to be 0.0525% and 0.000492% of the annual limit of 1 mSv/yr for the general public.

Under the accident conditions, the annual individual doses of residents were 0.0011%, 0.0023%, 0.0034%, and 0.0046% of the annual limit of 1 mSv/yr when carrying 5, 10, 15, and 20 casks.

Currently, the site of the interim storage facility has not been precisely determined, but a preliminary radiation assessment through sea transport resulted in a significantly lower than the limit. Combined scenario sea transport followed by land transport will be carried out in the next stage of study.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: SF, Interim Storage, Sea Transport, KORAD-21, Radiation

Corrosion Behavior of Neutron Absorber Borated Stainless Steel in Simulated Spent Fuel Pool Environment

Daehyeon Park¹, Yunju Lee¹, Junhyuk Ham¹, Seung Chang Yoo¹, Kiyoun Kim², Donghee Lee²,
Yongdeog Kim², and Ji Hyun Kim^{1,*}

¹Ulsan National Institute of Science and Technology (UNIST)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kimjh@unist.ac.kr

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

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

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

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

Acknowledgements

This work was financially supported by Korea Hydro & Nuclear Power Co., Ltd. This study contains the results obtained by using the equipment of UNIST Central Research Facilities (UCRF).

Keywords: Neutron Absorber, Borated Stainless Steel, Accelerated Corrosion Test, Pitting Corrosion, Dissolution

Efficient Radiation Shielding Evaluation for Spent Fuel Storage Facilities Using a Simple Model

Jae Hun Ko*, Young Hwan Choi, and Dong Gyu Lee

KONES Corp.

*kojh4@kones21.com

Radiation dose rates for spent fuel storage casks and storage facilities of them are typically calculated using Monte Carlo calculation codes. In particular, Monte Carlo computer code has the advantage of being able to analyze radiation transport very similar to the actual situation and accurately simulate complex structures.

However, to evaluate the radiation dose rate for models such as ISFSI (Independent Spent Fuel Storage Installation) with a lot of spent fuel storage casks using Monte Carlo computational techniques has a disadvantage that it takes considerable computational time. This is because the radiation dose rate from the cask located at the outermost part of the storage facility to hundreds of meters must be calculated. In addition, if a building is considered in addition to many storage casks, more analysis time is required. Therefore, it is necessary to improve the efficiency of the computational techniques in order to evaluate the radiation dose rate for the ISFSI using Monte Carlo computational codes.

The radiation dose rate evaluation of storage facilities using evaluation techniques for improving calculation efficiency is performed in the following steps. (1) simplified change in detailed analysis model for single storage cask, (2) create source term for the outermost side and top surface of the storage cask, (3) full modeling for storage facilities using casks with surface sources, (4) evaluation of radiation dose rate by distance corresponding to the dose rate limit.

Using this calculation method, the dose rate according to the distance was evaluated by assuming that the concrete storage cask (KORAD21C) and the horizontal storage module (NUHOMS-HSM) were stored in the storage facility. As a result of calculation, the distance to boundary of the radiation control area and restricted area of the storage facility is respectively 75 m / 530 m (KORAD21C case), and 20 m / 350 m (NUHOMS-HSM case).

Keywords: Efficient Radiation Shielding Evaluation, Monte Carlo Computer Code, Spent Fuel Storage Facilities, 2 Step Calculation Technique

3분과

고준위폐기물(Oral)

Corrosion Behavior of Potential Materials for Canister Fabricated Using 3D Printing Methods

Cheong Won Lee*, Young-Ho Lee, and Sung Chan Yoo

Korea Atomic Energy Research Institute (KAERI)

*cwlee2020@kaeri.re.kr

Spent nuclear fuels in Korea are temporarily stored at the nuclear power plant site and it is expected that will become saturated from 2031. Deep geological disposal in engineered barrier system (EBS) is one of the most important options for disposing spent nuclear fuel. The disposal canister is the first barrier that prevents leakage of nuclides in the spent nuclear fuel to the environment. Therefore, the corrosion behavior of the canister materials are significant factors in determining the overall disposal period. Oxygen-free copper is the most widely used material for disposal canisters, and manufacturing methods include forging, cold spray, and electro-deposition. In this study, corrosion behavior of materials that have the potential to replace oxygen-free copper manufactured using various 3D printing method were analyzed. As a result of electrochemical analysis of various materials such as copper manufactured by the Atmospheric Plasma Spray (APS) process and Inconel 718 manufactured by the Direct Energy Deposition (DED) process, the possibility of replacing oxygen-free copper was confirmed.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (2021M2E3A2041351) and the Institute for Korea Spent Nuclear Fuel and the National Research Foundation of Korea (2021M2E1A1085193).

Keywords: Spent Nuclear Fuel, Canister, Corrosion, 3D Printing, Electrochemical Analysis

Geochemical Simulation of U(VI) Sorption on Na-Montmorillonite: Effects of pH, Temperature, Competing Cations, U(VI) Concentration, and Carbonate Species

Jeonghwan Hwang¹, Weon Shik Han^{2,*}, Jung-Woo Kim¹, and Won Woo Yoon²

¹Korea Atomic Energy Research Institute (KAERI)

²Yonsei University

*hanw@yonsei.ac.kr

Montmorillonite plays a key role in engineered barrier systems in the high-level radioactive waste repository because of its large sorption capacity and high swelling pressure. However, the sorption capacity of montmorillonite can be largely varied dependent on the surrounding environments. This study conducted the batch simulation for U(VI) sorption on Na-montmorillonite by utilizing the cation exchange and surface complexation coupled (2SP-NE-SC/CE) model and evaluated the effects of physicochemical properties (i.e., pH, temperature, competing cations, U(VI) concentration, and carbonate species) on U(VI) sorption.

The simulation demonstrated that the U(VI) sorption was affected by physicochemical properties: the pH and temperature relate to aqueous U(VI) speciation, the competing cations relate to the cation exchange process and selectivity, the U(VI) concentration relates to saturation at sorption sites. For example, the K_d (L kg^{-1}) of Na-montmorillonite represented the largest values of $2.7 \times 10^5 \text{ L kg}^{-1}$ at neutral pH condition and had significantly decreased at acidic $\text{pH} < 3$, showing non-linear and diverse U(VI) sorption at the ranged pH from 2 to 11. Additionally, the U(VI) sorption on montmorillonite significantly decreased in presence of carbonate species. The U(VI) sorption for long-term in actual porewater chemistry and temperature of high-level radioactive waste repository represented that the sorption capacity of Na-montmorillonite was affected by various external properties such as concentration of competing cation, temperature, pH, and carbonate species. These results indicate that geochemical sorption capacity of bentonite should be evaluated by considering both geological and aquifer environments in the high-level radioactive waste repository.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: Simulation, Sorption, Montmorillonite, Uranium

Evaluation of Deep Geological Characteristics Using Deep Boreholes

Dae-Sung Cheon*, You Hong Kihm, Kwangmin Jin, Wonkyong Song, Seong Kon Lee, Sung Phil Hyun,
and Heejun Suk

Korea Institute of Geoscience and Mineral Resources (KIGAM)

*cds@kigam.re.kr

The depth of geological disposal of high-level radioactive waste (HLW) varies from country to country, but it is generally considered below 300 m underground. As one of the reliable methods to understand the geological characteristics of these deep areas, the site investigation through drilling is recommended. This paper deals with multidisciplinary research that evaluates the geological characteristics of the site using deep drilling. The deep drilling is 750 m, which is higher than the planned disposal depth. Prior to drilling, literature and surface geological surveys of the target area were conducted, and during drilling, real-time measurement of excavated information for obtaining drilling information, circulating water management and chemical composition through a closed system were monitored. After drilling, field tests such as geophysical borehole logging, deep groundwater sampling, constant pressure injection test, and hydraulic fracturing test were performed. Analysis of the recovered drilling core from a geological point of view such as age dating, rock formation and structural geological analysis, and from geochemical perspectives such as concentration of major/minor cationic elements, major anions, and trace elements along with the water quality parameters pH, DO, Ec, Eh, etc., from geothermal perspective such as thermal conductivity and coefficient of thermal expansion, from rock mechanical aspects such as physical and mechanical properties of intact rocks and joints, joint distribution, etc. Deep drilling has been completed with 2 holes for granite and 2 holes for sedimentary rocks, and further drilling for gneiss and sedimentary rocks is in progress.

Acknowledgements

This work was financially supported by the KIGAM's Basic Research Project (GP2020-002; 22-3115) funded by the Ministry of Science and ICT.

Keywords: Deep Geological Characteristics, Deep Borehole, High-level Radioactive Waste, Multidisciplinary Research, Site Investigation

Preliminary Experimental Results for Diffusion of Sr, Sm, and Eu in a Compacted Ca-Bentonite

Nak Kyu Kim*, Chung-Kyun Park, Jae-Kwang Lee, Seonggyu Choi, and Jang-Soon Kwon

Korea Atomic Energy Research Institute (KAERI)

*kimnk@kaeri.re.kr

The deep geological disposal system is aimed to permanently isolate the high-level radioactive waste from the biosphere through a multi-barrier system composed of engineered and natural barriers. The buffer material used for the engineered barrier should have the performance to prevent and retard the migration of radionuclides to the outside of the deep disposal facility when radionuclides are released from a disposal canister by infiltration of groundwater after a long period of time. When the hydraulic conductivity of compacted bentonite is sufficiently low, the migration of radionuclides released into the surrounding rock can be significantly reduced since they are sorbed to bentonite during the diffusion process. Therefore, an investigation on diffusion of radionuclides in compacted bentonite is a fundamental task to obtain essential data for the safety assessment of the deep geological disposal system.

The migration of radionuclides by diffusion can be evaluated by diffusion coefficient. In order to obtain the apparent diffusion coefficients of Sr, Sm, and Eu in compacted Ca-bentonite (1.6 g/cm^3) a through-diffusion experiment have been carrying out. A cylindrical apparatus consists of a source cell with an appropriate concentration of radionuclides and diffusion cell filled with radionuclide free solution where the concentration is gradually increased by diffusion of radionuclides. The compacted bentonite was installed between the both cells. The sample used for the experiment is a Ca-type bentonite named Bentonil-WRK, and the diffusion experiment was performed under an oxidizing condition using a synthetic groundwater simulating KURT groundwater composition.

The diffusion experiment will be terminated when an increasing rate of concentration of nuclides in the diffusion cell becomes constant over time. The concentration change with regard to the geochemical characteristic of the nuclide may appear to be apparently slow. In this study, the experimental results of the through-diffusion test of Sr, Sm, and Eu in the initial stage (~4 months) were presented. Through the results of the initial stage, the period of the through-diffusion experiment can be rearranged and also it is expected that the initial results provide the qualitative and quantitative diffusion properties of each nuclide.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085202).

Keywords: Diffusion Coefficient, Bentonite, Through-diffusion Experiment, Strontium, Samarium, Europium

Preliminary Study for Estimating the Effect of Earthquake-induced Stress on Hydraulic Properties of Fractures

Yong-Min Kim* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*kym@kaeri.re.kr

With the increase of temporarily-stored radioactive waste in Korea, the disposal of radioactive waste in a deep geological repository, which is located in crystalline rock at a depth of hundreds of meters below the ground level, has received great attention nowadays. To ensure the permanent isolation of radionuclides from the human and surrounding ecosystems, the safety assessment for the high-level radioactive waste disposal facilities is essential. For the reliable safety assessment of fractured rock, it is especially important to input proper hydraulic properties of fractures such as aperture and hydraulic conductivity, which can directly affect the fluid flow and radionuclide transport.

Meanwhile, it has become important to consider sudden fault behavior caused by an earthquake with the recent occurrence of high-intensity earthquakes in the Korean Peninsula. The sudden fault behavior can induce the changes of the hydraulic properties of fractures. Since the changes of the hydraulic properties directly affects to the radionuclide transport in the fractured rock, it is important to estimate the effect of earthquake-induced stress change on hydraulic properties of fractures in the perspective of long-term safety assessment.

In this study, the effect of an earthquake on the hydraulic properties of fractures was explored by a numerical approach. The static Coulomb stress change after the earthquake was calculated using software 'Coulomb 3' developed by United States Geological Survey (USGS) with the assumption for several mechanical properties such as Young's modulus, Poisson's ratio and effective coefficient of friction. The final stress after earthquake occurrence was calculated as the sum of the initial stress and the stress change. Thereafter, the normalized transmissivity of fracture after the earthquake was calculated using the final stress from the stress-transmissivity relationship. Using the methodology for calculating fracture transmissivity change induced by the earthquake developed in this study, the effect of several factors, such as the earthquake magnitude and the distance between fracture and epicenter, was additionally explored. The newly developed methodology will be applied to the process-based total system performance assessment framework (APro) being developed by KAERI, and this study is expected to be helpful for the safety assessment considering long-term evolution phenomena including earthquakes.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: Earthquake, Fracture, Transmissivity, Aperture, Coulomb Stress, APro

Inversion of Barcelona Basic Model Parameters of Kunigel-V1

Yeongju Kim¹, Hoonyoung Jeong^{1,*}, Changsoo Lee², and Minseop Kim²

¹Seoul National University (SNU)

²Korea Atomic Energy Research Institute (KAERI)

*hoonyoung.jeong@snu.ac.kr

For safety assessment of a high-level radioactive waste disposal system, it is important to predict and analyze the coupled thermo-hydro-mechanical (THM) behaviors of bentonite, which is a buffer candidate material in the engineered barrier system. The Barcelona Basic Model (BBM) is a constitutive model to describe the geomechanical behaviors of partially saturated soils. Complicated tests are required to directly measure BBM parameters of bentonite. In this study, we demonstrate that probable BBM parameters can be sought by calibrating the BBM parameters to match simulation results to observed ones for two kinds of simple tests (swelling pressure test and free swelling test) instead of the complicated direct tests. In the swelling pressure test and free swelling test that were conducted by Japan Atomic Energy Agency (JAEA), water was injected into constrained and unconstrained bentonite core samples, and then swelling pressure and displacements were measured, respectively. We find optimal BBM parameters using a quasi-Newton optimization method that reproduce the observed swelling pressures and displacements in hydro-mechanical simulations. The optimal BBM parameters that are sought in the inversion process can be used to predict the THM behaviors of bentonite barriers in a high-level radioactive waste disposal system.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (2021M2E1A1085193).

Keywords: Inversion, Kunigel-V1, Bentonite, Barcelona Basic Model, Hydro-Mechanical Simulation, Safety Assessment

Analyses on Thermal Stability of the Disposal Systems for High Burn-up PWR Spent Nuclear Fuel in Korea

Gisu Lee, Myeonggi Cha, Namhyeong Kim, and HangJin Jo*

Pohang University of Science and Technology (POSTECH)

*jhj04@postech.ac.kr

Recently, the deep geological disposal system isolating a spent nuclear fuel (SNF) is considered a disposal method of high-level radioactive waste for the safety of humans or the natural environment. The one of important requirements for maintaining the thermal stability of these systems is that the temperature of the buffer does not exceed 100°C even though the decay heat is emitted from high-level radioactive wastes loaded in the disposal container. In 2007, a deep geological disposal system based on the Swedish disposal concept was developed for the SNF in Korea. To respond to the development process, the thermal stability of the deep geological disposal system developed for the disposal of domestic pressurized light water reactor (PWR) SNFs with discharged burn-up of 55 GWD/MTU was evaluated in 2019. The thing is that the recent fuel activity is pursuing to operate further high burn-up fuel conditions, and it leads to emergency core cooling system (ECCS) revision for extending the license for up to 60 or more than 60 GWD/MTU in the world. In this regard, this study evaluates numerically the thermal stability of the deep geological disposal system for the high burn-up PWR SNF having large decay heat compared to previous conditions for two different length disposal containers classified according to the length of PWR SNFs discharged from domestic nuclear power plants. A finite element analysis using a computational program was used to evaluate the thermal design requirements. Results show that both types of disposal containers would increase the temperature which reduces or fails to meet the safety margin of the disposal system. This study suggests that the design of the previous disposal system is needed to be further developed for the high burn-up PWR SNF which would be used in future nuclear power plant systems.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Foundation of Nuclear Safety (KOFONS) grant funded by the Korea government (Nuclear Safety and Security Commission, NSSC) (No. 2109092-0222-SB120).

Keywords: Spent Nuclear Fuel, Deep Geological Disposal, Thermal Stability, Design Requirements, High Burn-up

~~Methodologies to Establish Site Specific FEP for High-Level Radioactive Waste Repository in Korea~~

~~Donghoon Baek, Kibeom Son, Jaehyeon Yang, Haeram Jung, Karyoung Choi, Gyunyoung Heo,
Hyungdae Kim, and Kunok Chang*~~

~~Kyung Hee University~~

~~*kunok.chang@khu.ac.kr~~

Withdrawal

3분과

고준위폐기물(Poster)

Characteristics of AE Wave Propagation Through Concrete Silo of Gyeongju Radioactive Waste Repository Under the Effects of Groundwater and Temperature

Jin-Seop Kim^{1,*}, Hang-Lo Lee², Chang-Ho Hong¹, and Ji-Won Kim¹

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Expressway Corporation

*kjs@kaeri.re.kr

The structural integrity of concrete silos is important from the perspective of long-term operation of radioactive waste repository. Recently, the application of acoustic emission (AE) is considered as a promising technology for the systematic real-time health monitoring of concrete-like brittle material. In this study, the characteristics of AE wave propagation through concrete silo of Gyeongju radioactive waste repository were evaluated under the effects of groundwater and temperature for the quantitative damage assessment. The attenuation coefficients and absolute energies of AE waves were measured for the temperature cases of 15, 45, 75°C under dry and saturated concrete specimens, which were manufactured based on the concrete mix same as that of Gyeongju concrete silo. The geometric spreading and material loss were taken into account with regard to the wave attenuation coefficient. The attenuation coefficient shows a decreasing pattern with temperature rise for both dry and saturated specimens. The AE waves in saturated condition attenuate faster than those in dry condition. It is found that the effect of water content has a greater impact on the wave attenuation than the temperature. The results from this study will be used as valuable information for estimating the quantitative damage at the location micro-cracks are generated rather than the AE sensor location.

Keywords: Radioactive Waste Disposal, Wave Propagation, Wave Attenuation, AE, Concrete, Gyeongju

Factors Controlling Uranium Level in Granitic Aquifer

YeoJin Ju^{1,*}, Vikram Thapar², and Ji-hun Ryu¹

¹Korea Atomic Energy Research Institute (KAERI)

²Cornell University

*yju@kaeri.re.kr

Uranium (U) is hazardous material can cause chemical and radiological toxicity, e.g., kidney toxicity and health effects associated with radiation. In Korea, where shallow weathered granitic aquifers are ubiquitous, several previous studies have reported high level of radioactivity in shallow groundwater. This eventually led to the closure of 60 out of 4,140 groundwater production wells in South Korea. Here, we examined aquifers currently dedicated for drinking water supply and investigated the 11,225 dataset of 103 environmental parameters. This dataset includes 80 physical parameters associated with the hydraulic system and 23 chemical parameters associated with water-rock interactions. Among hydraulic parameters, coarse loamy texture in subsoil displayed a notable relation with U concentration level, implied it is controlling the leaching of U from host rocks. Fluorine (F), is one of major products from water-rock interaction in granitic aquifer, exhibited high correlation with U concentration distribution. Positive relation of F concentration with uranium level suggested the dissolved U originated from groundwater interacted with granites. Conclusively, we found that infiltration capacity of soil layers and (2) aqueous speciation in groundwater formulated by interaction of groundwater with local solids, played important role for U concentration in granitic aquifer.

Acknowledgements

The Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by MSIT (No. 2021M2E1A1085186).

Keywords: Uranium, Granite, Groundwater, AI

Spectroscopic Study of Interaction Between U(VI) and Natural Organic Matters in Domestic Groundwater

Euo Chang Jung*, Min-Hoon Baik, Dawoon Jeong, and Ji-Hun Ryu

Korea Atomic Energy Research Institute (KAERI)

*ecjung@kaeri.re.kr

Time-resolved laser fluorescence spectroscopy (TRLFS) and excitation-emission matrix (EEM) spectroscopy were used to study the interaction of U(VI) and natural organic matters (NOMs) in groundwater. Various types of groundwaters (DB-1, DB-3 from KURT site and OB-1, OB-3 from a U deposit in Ogcheon metamorphic belt) were used as samples. Pulsed Nd-YAG laser at 266 nm (Continuum Minilite) was used as the light source of TRLFS. The laser pulse energy of 1.0 mJ was fixed for all measurements. The luminescence spectrum was recorded using a gated intensified charge-coupled device (Andor, DH-720/18U03 iStar 720D) attached to the spectrograph (Andor, SR-303i). EEM spectra were measured using a spectrofluorometer (Horiba Scientific, Aqualog) equipped with a 150 W ozone-free xenon arc lamp. Excitation spectra were recorded by scanning the excitation wavelength from 200 to 500 nm. Emission spectra were measured using a CCD in the wavelength range of 242–823 nm.

In the case of the recently collected DB-1 samples, it was observed that the U and NOM quantities decreased compared to the samples collected before 2016. For some DB-1 samples, the amount of dissolved organic carbon indicating the presence of NOM was significantly reduced, and changes consistent with this phenomenon were observed in the EEM spectrum. The time-resolved luminescence characteristics (peak wavelengths and lifetime) of U(VI) in the DB-1 samples agree well with those of $\text{Ca}_2\text{UO}_2(\text{CO}_3)_3(\text{aq})$. This U(VI) species remains stable, even in samples taken five years ago. The estimated amounts of U and NOM from the spectroscopic data of DB-3, OB-1, and OB-3 samples are relatively low compared to DB-1 samples. When a known amount of U(VI) was mixed in each groundwater, the time-resolved luminescence spectrum exhibited a characteristic spectral shape different from the expected luminescence intensity. This phenomenon is presumed to be due to the interaction between U(VI) and NOM in groundwater. The results of this study suggest that the chemical speciation of NOM as well as U(VI) is required to understand U behavior in groundwater.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No 2021M2E1A1085186).

Keywords: Time-resolved Laser Fluorescence Spectroscopy (TRLFS), Excitation-emission Matrix (EEM), Groundwater, U(VI), Natural Organic Matter (NOM)

Removal of Aqueous Uranium by Indigenous Bacteria Under Conditions Relevant for Deep Geological Repository of Nuclear Waste

Dawoon Jeong¹, Min Hoon Baik¹, Euo Chang Jung¹, Myoung-Soo Ko², Wooyong Um³, and Ji-Hun Ryu^{1,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Kangwon National University (KNU)

³Pohang University of Science and Technology (POSTECH)

*jryu@kaeri.re.kr

Spent nuclear fuel (SNF) is the main source of high-level radioactive wastes (HLWs), which contains approximately 96% of uranium (U). For the safe disposal of the HLWs, the SNF is packed in canisters of cast iron and copper, and then is emplaced within 500 m of host rock surrounded by compacted bentonite clay buffer for at least 100,000 years. However, in case of the failure of the multi-barrier disposal system, U might be migrated through the rock fractures and groundwater, eventually, it could reach to the biosphere. Since the dissolved U interacts with indigenous bacteria under natural and engineered environments over the long storage periods of geologic disposal, it is important to understand the characteristics of U-microbe interactions under the geochemical conditions. In particular, a few of bacteria, i.e., sulfate-reducing bacteria (SRB), are able to reduce soluble U(VI) into insoluble U(IV) under anaerobic conditions by using their metabolisms, resulting in the immobilization of U.

In this study, the aqueous U(VI) removal performance and change in bacterial community in response to the indigenous bacteria were investigated to understand the interactions of U-microbe under anaerobic conditions. Three different indigenous bacteria obtained from different depths of granitic groundwater (S1: 44–60 m, S2: 92–116 m, and S3: 234–244 m) were used for the reduction of U(VI)_{aq}. After the anaerobic reaction of 24 weeks, the changes in bacterial community structure in response to the seeding indigenous bacteria were observed by high-throughput 16S rDNA gene sequencing analysis. The highest uranium removal efficiency of 57.8% was obtained in S3 sample, and followed by S2 (43.1%) and S1 (37.7%). Interestingly, SRB capable of reducing U(VI) greatly increased from 4.8% to 44.1% in S3 sample. Among the SRB identified, *Thermodesulfovibrio yellowstonii* played a key role on the removal of U(VI)_{aq}. Transmission electron microscopy (TEM) analysis showed that the d-spacing of precipitates observed in this study was identical with that of uraninite (UO₂). This study presents the potential of U(VI)_{aq} removal by indigenous bacteria under deep geological environment.

Keywords: Uranium, Bioreduction, Indigenous Bacteria, Deep Geological Repository (DGR), Illumina Miseq Analysis, Sulfate-reducing Bacteria (SRB)

Scenario Development Methods for Deep Geological Repository:

(2) Case Study on Disposal Facility for the Spent Nuclear Fuel

Kibeom Son, Karyoung Choi, Jaehyeon Yang, Haeram Jeong, Donghoon Baek, Hyungdae Kim,

Kunok Chang, and Gyungyoung Heo*

Kyung Hee University

*gheo@khu.ac.kr

In Korea, research on the development of safety case, including the safety assessment of disposal facility for the spent nuclear fuel, is being conducted for long-term management planning. The safety assessment procedure on disposal facility for the spent nuclear fuel heavily involves creating scenarios in which radioactive materials from the repository reach the human biosphere by combining Features, Events and Processes (FEP) that describe processes or events occurring around the disposal area. Meanwhile, the general guidelines provided by the IAEA or top-tier regulatory requirements addressed by each country do not mention detailed methods of ‘how to develop scenarios by combining individual FEPs’. For this reason, the overall frameworks of developing scenarios are almost similar, but their details are quite different depending on situation. Therefore, in order to follow up and clearly analyze the methods of how to develop scenarios, it is necessary to understand and compare case studies performed by each institution. In the previous companion paper entitled ‘Research Status and Trends’, the characteristics and advantages/disadvantages of representative scenario development methods were described. In this paper, which is a next series of the companion papers, we investigate and review with a focus on details of scenario development methods officially documented. In particular, we summarize some cases for the most commonly utilized methods, which are categorized as the ‘systematic method’, and this method is addressed by Process Influence Diagram (PID) and Rock Engineering System (RES). The lessons-learned and insight of these approaches can be used to develop the scenarios for enhanced Korean disposal facility for the spent nuclear fuel in the future.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (Ministry of Trade, Industry and Energy (MOTIE)) (No. 2021040101003C).

Keywords: Features, Events and Processes (FEP), Scenario Development, Deep Geological Repository, Spent Nuclear Fuel

Digital Analysis Technique for Aperture and Contact Area of Fractures

Yong-Ki Lee*, Chae-Soon Choi, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*ykle12@kaeri.re.kr

The hydro-mechanical behavior of rock mass in natural barriers is a critical factor of interest, and it is mainly determined by the characteristics of the fractures distributed in the rock mass. In particular, the aperture and contact area of the fractures are important parameters directly related to the fluid flow and significantly influence the hydro-mechanical behavior of natural barriers. Therefore, it is necessary to analyze the aperture and contact area of fractures distributed in potential disposal sites to examine the long-term evolution of the natural barriers. This study aims to propose a new technique for analyzing the aperture and contact area using the natural fractures in KURT (KAERI Underground Research Tunnel), an underground research facility for the deep geological disposal of high-level radioactive waste.

The proposed technique consists of a matching algorithm for the three-dimensional point cloud of the upper and lower fracture surfaces and a normal deformation algorithm that considers the fracture normal stiffness. In the matching process of upper and lower fracture surfaces, digital images obtained from compression tests with pressure films are used as input data. First, for the primary matching of the upper and lower fracture surfaces, an iterative closest point (ICP) algorithm is applied in which rotation and translation are performed to minimize the distance error. Second, an algorithm for rotation about the x, y, and z axes and translation in the normal direction is applied so that the contact area of the point cloud is as consistent as possible with the pressure film image. Finally, by applying the normal deformation algorithm considering the fracture normal stiffness, the aperture and contact area of the fracture according to the applied normal stress are derived. The applicability of the proposed technique was validated using 12 natural fractures sampled from KURT, and it was confirmed that the initial apertures were derived similarly to the empirical equation proposed in the previous study. Therefore, it was judged that the distribution of apertures and contact areas according to applied normal stress for laboratory-scale fractures could be derived through the technique proposed in this study.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and the National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No.2021M2E1A1085200).

Keywords: Aperture, Contact area, Fracture, Digital Analysis, Natural Barrier

Scenario Development Methods for Deep Geological Repository:

(1) Research Status and Trends

Karyoung Choi, Kibeom Son, Jaehyeon Yang, Haeram Jeong, Donghoon Baek, Hyungdae Kim,

Kunok Chang, and Gyunyoung Heo*

Kyung Hee University

*gheo@khu.ac.kr

Safety assessment is important for the radioactive waste repositories, and several methods are used to develop scenarios for the management of radioactive waste. The intent of the use of these scenarios is to show how the radio nuclides release can affect the safety of disposal system. It plays an essential role of providing scientific and technical information for performance assessment of safety functions. As important as scenario is, numerous studies for their own scenario development have been conducted in many countries.

Scenario development methodology is basically divided into four categories: (1) judgmental, (2) fault/event-tree analysis, (3) simulation, and (4) systematic. Under numerous research, these methods have been developed in ways to strengthen the advantages and make up for the weakness. However, it was hard to find any judgmental or fault/event-tree analysis approach in recent safety assessments since they are not well-systemized and difficult to cover all scenarios. Simulation and systematic approaches are used broadly for their convenience of analyzing needed scenarios. Furthermore, several new methodologies, Process Influence Diagram (PID)/Rock Engineering System (RES)/Hybrid, were developed to reinforce the systematic approach in recent studies.

Currently, a government project related to the disposal of spent nuclear fuel is in progress in Korea, and the scenario development for safety case is one of the important tasks. Therefore, it is necessary to identify the characteristics and strengths and weaknesses of the latest scenario development and analysis methods to create a unique methodology for Korea.

In this paper, the existing methodologies and cases will be introduced, and the considerations for future scenario development will be summarized by considering those used in the nuclear field other than repository issues. Systematic approach, which is the mostly commonly used method, will be introduced in detail with its use in other countries at the subsequent companion paper entitled 'Case Study for a Disposal Facility for the Spent Nuclear Fuel'.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (Ministry of Trade, Industry and Energy (MOTIE)) (No. 2021040101003C).

Keywords: Features, Events and Processes (FEP), Scenario Development, Deep Geological Repository

Preliminary Study on Accelerated Life Test of Piezoelectric Sensor

Hyun-Joong Hwang¹, Changhee Park¹, Gye-Chun Cho^{1,*}, and Chang-Ho Hong²

¹Korea Advanced Institute of Science and Technology (KAIST)

²Korea Atomic Energy Research Institute (KAERI)

*gyechun@kaist.ac.kr

The high-level waste disposal system is an underground structure exposed to complex environmental conditions such as high temperature, radiation, and groundwater. The high-level waste disposal causes structural cracks and deterioration over time. However, since the high-level waste disposal system is a structure that should be operated for a very long time, developing a high-durability monitoring sensor to detect cracks and deterioration is essential. The durability of the sensor can be evaluated by predicting the expected life through the accelerated life test, one of the reliability qualification tests. The most important factor in the accelerated life test design is setting the harsh stress level. This study figured out the harsh stress level of the piezoelectric sensor, which is commonly used for underground structure monitoring. It is possible to determine the appropriate stress level for the accelerated life test by investigating the harsh stress level for the temperature factor. It will contribute to more accurate life expectancy prediction.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (2022M2E3A3015608) funded by the Minister of Science and ICT.

Keywords: High-level Waste Disposal, Piezoelectric Sensor, Harsh Stress Level, Accelerated Life Test

Evaluation of Fracture Filling Minerals and Aperture Distribution of Fractured Granite

Bong-Ju Kim, Ki-Jong Jang, Chung-kyun Park, and Jang-Soon Kwon*

Korea Atomic Energy Research Institute (KAERI)

*jskwon@kaeri.re.kr

In this study, we evaluated fracture filling minerals and aperture distribution along the fracture surfaces under the controlled conditions. The fractured granite block which has a single natural fracture of 1 m scale was sampled in a domestic quarry (Iksan), which groundwater had been flowed through. This rock has an interconnected porosity of 0.36 with the specific gravity of 2.57. The experimental setup with the granite block with dimensions of 100×60×60 (cm). The fracture is sealed with rock silicone rubbers when it intersects the outer surfaces of the block and the outer surfaces are coated with the silicone to prevent loss of water by evaporation. Nine boreholes were drilled of orthogonal direction at the fracture surface. A flow of de-ionized water through the fracture between pairs of boreholes was initiated and the pressure required to maintain a steady flow was measured. In additions, fracture filling minerals were sampled and examined by mineralogical and chemical analyses. There are phyllosilicate minerals such as illite, kaolinite, and chlorite including calcite, which are fracture filling minerals. The illite and kaolinite usually coexist in the fracture, where their content ratio is different according to which mineral is predominant. For the evaluation of fracture, surface was divided into an imaginary matrix of 20×20 sub-squares as schematically. The calculated results are expressed as a two dimensional contour and a three dimensional surface plot for the aperture distribution in the fracture. The aperture value is distributed between 0.075 and 0.114 mm and the mean aperture value is 0.082 mm. The fracture volume is about 49 ml. These results will be very useful for the evaluation of environmental factor affecting the nuclides migration and retardation.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (No. 2021M2E1A108 5202).

Keywords: Fracture Filling Minerals, Aperture Distribution, Fracture, Granite, Migration

Comparison of Two Disposal Concepts for CANDU Spent Nuclear Fuels in Korea

Heui-Joo Choi*, Jong Youl Lee, Dong-Keun Cho, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*hjchoi@kaeri.re.kr

It is expected that around 576,000 bundles of CANDU spent nuclear fuels (SNF) will be generated from the four CANDU reactors located at the Wolsong site, according to the 2nd National Plan for the management of High-Level radioactive Waste (HLW). The CANDU SNFs are currently stored at the dry storage facilities at the Wolsong site. The authors proposed KRS⁺ geological disposal system consisting of two different concepts, Swedish KBS-3V type and Canadian NWMO type, for the final management of CANDU SNF. Both the concepts were designed based on the geological data obtained from the KURT (KAERI Underground Research Tunnel). The NWMO type is an in-room horizontal placement method.

In this study, we try to determine the reference concept among the two proposed concepts at 500 meters below the ground surface. Assuming 10,000 tU of CANDU SNF and the KURT site, we design two engineered barrier systems, that is disposal canisters and buffers. The copper disposal canister is designed with a copper thickness of 10 mm based on a cold spray coating technique for both the disposal concepts. The domestic Ca-bentonite is used for the compact bentonite buffer with dry density of 1.6 g/cm³. Two concepts are compared in terms of safety, economics of the engineered barriers, and environment-friendliness. Because the same amounts of CANDU SNF are disposed of at the same depth, the differences in the disposal area are neglected.

For the comparison in terms of safety, the corrosion lifetimes of the disposal canisters of two disposal systems are quantitatively calculated, and the capacities for retarding radionuclide releases of the compacted bentonite buffers are assessed. A computer tool developed by the authors is used in order to assess the lifetime of a disposal canister. In this study, the case that corrosion of a copper canister by sulfide from groundwater through intact buffer is analyzed. The sulfide concentration in groundwater is assumed to be 3 ppm. The most important safety function of buffer is to retard the radionuclide release. Twelve long-lived radionuclides are selected to compare the capacities for retarding the radionuclide transport through the buffer using an analytical solution. The retention time by an engineered barrier consisting of a disposal canister and a buffer is compared with twenty times the half-life of each radionuclide for both the disposal systems. The selected reference concept will be compared with the alternative geological concepts through a further study.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: CANDU Spent Nuclear Fuel, Disposal Concepts, KBS-3V, NWMO Type, Copper Disposal Canister, Bentonite Buffer

Analysis and Design of Down Ender and Dismantler for Spent Fuel Assembly Dismantling in Head-End

Younghwan Kim*, SunSeok Hong, Seckmin Hong, JuHo Lee, JaeWon Lee, and ChangHwa Lee

Korea Atomic Energy Research Institute (KAERI)

*Yhkim3@kaeri.re.kr

Dry head end process is developing for pyro-processing at KAERI (Korea Atomic Energy Research Institute). Dry processes, which include disassembling, mechanical decladding, vol-oxidation, blending, compaction, and sintering shall be performed in advance as the head-end process of pyro-processing. Also, for the operation of the head-end process, the design of the connecting systems between the down ender and the dismantling process is required. The disassembling process includes apparatus for down ender, dismantling of the SF (Spent Fuel) assembly (16×16 PWR), rod extraction, and cutting of extracted spent fuel rods. The disassembling process has four-unit apparatus, which comprises of a down ender that brings the assembly from a vertical position to a horizontal position, a dismantler to remove the upper and bottom nozzles of the spent fuel assembly, an extractor to extract the spent fuel rods from the assembly, and a cutter to cut the extracted spent fuel rods as a final step to transfer the rod-cuts to the mechanical decladding process. An important goal of dismantling process is the disassembling of a spent nuclear fuel assembly for the subsequent extraction process. In order to design the down ender and dismantler, these systems were analyzed and designed, also concept on the interference tools between down ender and dismantler were considered by using the solid works tool.

Acknowledgements

This study was supported by the National Research Foundation of Korea grant 2021M2E3A1040059 funded by the Korean government.

Keywords: Head-end, Disassembling, Dismantler, Down-ender, Spent, Fuel, Assembly

A Natural Analogue Study Using Uranium Deposits: Characterization of a Study Site in the Ogcheon Metamorphic Belt

Min-Hoon Baik*, Dawoon Jeong, YeoJin Ju, Euo Chang Jung, and Ji-Hun Ryu

Korea Atomic Energy Research Institute (KAERI)

*mhbaik@kaeri.re.kr

In south Korea, most of uranium deposits are distributed in the Ogcheon belt, which is one of two late Precambrian to Paleozoic fold belts (the Imjingang and Ogcheon belts). A study site of the Ogcheon metamorphic belt (OMB) in Hoenam-myun, Boeun-gun was selected for the natural analogue study by preliminary site investigation for several candidate study sites. Three boreholes were drilled in the site and some rock cores and groundwater samples were taken from the boreholes. Various analytical studies for the samples are now being performed. Thus, in this study, various basic characteristics of the study site such as occurrence, geological, mineralogical, and chemical properties were investigated for a future study. Base rocks containing uranium in the OMB are usually black slate and coaly slate. Coaly slate usually shows a higher content of uranium and larger grain size of uranium than black slate. Uranium minerals found in the OMB are uraninite, uranothorite, brannerite, ekanite, coffinite, francevillite, uranophane, autunite, and torbernite depending on the base rock types. Uranothorite is abundant in black slate whereas uraninite is mostly abundant in coaly slate. Chemical compositions of the solid and groundwater samples from the study site were also analyzed by using ICP-MS/OES (Inductively Coupled Plasma Mass Spectrometry) and XRF (X-ray Fluorescence). This will contribute to determine uranium minerals in the solid samples and uranium speciation in the groundwater. The results of this study will contribute to performing future natural analogue studies in domestic uranium deposits and provide basic information and knowledge for understanding long-term geochemical behaviors of radionuclides in a high-level radioactive repository.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085186).

Keywords: Natural Analogue Study, Uranium Deposit, Ogcheon Metamorphic Belt, Black Slate, Coaly Slate

Vitrification of Uranium Oxide in Spent Nuclear Fuel Using Iron Phosphate Glass

Cheong Won Lee¹ and Jong Heo^{2,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Pohang University of Science and Technology (POSTECH)

*jheo@postech.ac.kr

Spent nuclear fuels are temporarily stored in nuclear power plant site. When a problem such as cracking of spent nuclear fuel assembly or cladding occurs or uranium that has not been separated during the reprocessing remains, it is necessary to treat it. The borosilicate glasses have been considered to vitrify whole spent nuclear fuel assembly. However, a large amount of Pb addition was necessary to oxidize metals in assembly to make them suitable for oxide glass vitrification. Furthermore, these borosilicate glasses need to be melted at high temperatures ($> 1,400^{\circ}\text{C}$) when UO_2 content is more than 20wt%. Iron phosphate glasses can be melted at a relatively low temperature ($< 1,300^{\circ}\text{C}$) even with a similar UO_2 addition. A composition of iron phosphate glass for immobilization of uranium oxide has been developed. The glasses have glass transition temperatures of $\sim 555^{\circ}\text{C}$ that are high enough to maintain its phase stability in geological repositories. The waste loading of UO_2 in the glass is $\sim 33.73\text{wt}\%$. Normalized elemental releases from the product consistency test were well below the US regulation of 2 g/m^2 . Nuclear criticality safety and heat generation in deep geological repositories were calculated using MCNP and computational fluid dynamics simulation, respectively. The glass had effective neutron multiplication factor (k_{eff}) of 0.755, which is smaller than the nuclear-criticality safety regulation of 0.95. Surface temperature of the disposal canister is expected to lower than the limit temperature ($< 100^{\circ}\text{C}$). Most of the U in the glass is in the 4+state, which is more chemically durable than the 6+state. As a result of long-term dissolution experiment, chemically-durable uranium pyrophosphate (UP_2O_7) crystals were formed.

Keywords: Spent Nuclear Fuel, Uranium Oxide, Iron Phosphate Glass, Vitrification, Waste Form

Development of FEP & Scenario Integrated Information Processing Platform for Long-Term Evolution of Spent Fuel Disposal System

Sang hun Jeon¹, Sunggyun Jang¹, Jeong Hyun Lee¹, Jun Ki An¹, Jin Ryu¹, Si On Kim¹, and Won-Seok Kim^{1,2,*}

¹NCsquare Co.

²Pohang University of Science and Technology (POSTECH)

*wkim23@ncsquare.co.kr

CYPRUS is a web-based waste disposal research comprehensive information management program developed by the Korea Atomic Energy Research Institute over three years from 2004. This program is stored as existing quality assurance documents and data, and the research results can be viewed at any time. In addition, it helps to perform all series of tasks related to the safety evaluation study of the repository in accordance with the quality assurance system. In the future, it is necessary to improve the user convenience by clarifying the relationship between FEP and scenarios and upgrading output functions such as visualization and automatic report generation. This purpose of this study is to research and develop the advanced program of CYPRUS.

This study is based on building FEP, DIM and scenario databases. It is necessary to develop an algorithm to analyze and visualize the FEP, DIM and scenario relationship. This project is an integrated information processing platform for DB management and visualization considering user convenience.

The first development goal is to build long-term evolutionary FEP, DIM, and scenarios as a database. The linkage by FEP item was designed in consideration of convenience by using a mixed delimiter of letters and numbers. This design provides information on detailed interactions and impacts between FEP items. Scenario data lists a series of events and characteristic change information for performance evaluation in chronological order. In addition, it includes information on FEP occurrence and mutual nutrition by period, and information on whether or not the repository performance is satisfied by item.

The second development goal is to realize the relationship analysis and visualization function of FEP and scenario based on network analysis technique. Based on DIM, this function analyzes and visualizes interactions between FEPs in the same way as PID, RES, etc. In addition, this function analyzes FEP and DIM using network analysis technique and visualizes it as a diagram.

The developed platform will be used to construct and visualize the FEP DB covering research results in various disposal research fields, to analyze and visualize the relationship between core FEP and scenarios, and finally to construct scenarios and calculation cases that are the evaluation target of the comprehensive performance evaluation model. In addition, it is expected to support the knowledge exchange of experts based on the FEP and scenario integrated information processing platform, and to utilize the platform itself as a part of the knowledge transfer system for knowledge preservation.

Acknowledgements

This work was supported by Korea Atomic Energy Research Institute (KAERI).

Keywords: Spent Fuel Disposal System, Integrated Information Processing Platform, Convenience, Visualization

Two-Phase Flow Analysis in the Full-Scale Emplacement Experiment at Mont Terri Underground Rock Laboratory

Taehyun Kim¹, Chan-Hee Park^{2,*}, Changsoo Lee¹, and Jin-Seop Kim¹

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

*chanhee.park@kigam.re.kr

In high-level radioactive waste disposal, a high temperature is generated from the canister containing the waste in the engineered barrier, while groundwater flows into the buffer system from the host rock. The temperature increase and groundwater inflow result in the water phase change and saturation variation. Saturation change is related to the thermal conductivity of buffer material; hence the phase change and saturation strongly interact with the temperature evolution. The complex coupled behavior affects the stability of the whole disposal system, and the security of the repository is critical to human-being life.

However, it is difficult to predict the long-term coupled behavior in the disposal system due to the considerable field-test scale, and therefore a numerical simulation is a suitable method having repeatability and cost-effectiveness. DECOVALEX is an international cooperating project for developing numerical methods and models for thermo-hydro-mechanical-chemical (THMC) interaction. DECOVALEX has a four-year cycle with various topics.

At the current phase, Task C aims to simulate the full-scale emplacement (FE) experiment performed at Mont Terri underground rock laboratory. Nine research groups are participating in the task, and among them, KAERI simulates the experiment using OGS-FLAC. The simulator combines OpenGeoSys for TH simulation and FLAC3D for M simulation.

Through the benchmark simulation, we verified OGS-FLAC for the two-phase flow analysis in the disposal system and finally modeled the FE experiment with a three-dimensional grid. We performed a simple sensitivity analysis to investigate the effect of input parameters on the two-phase flow system and confirmed that the compressibility and permeability affected the flow behavior. We also compared the simulation results to the field data and obtained well-matched results from a series of simulation.

Acknowledgements

This work was supported by a National Research Foundation (NRF) grand funded by the Korean government (MSIP) (No.2020M2C9A1062949 and No. 2021M2E3A2041312) and the Basic Research Project of the Korea Institute of Geoscience and Mineral Resources (KIGAM, GP2020-010) funded by the Ministry of Science and ICT, Korea.

Keywords: Two-phase Flow, FE Experiment, DECOVALEX, OGS-FLAC

Geochemical Behavior of Cesium and Cobalt in a Near-Surface and Deep Geological Disposal Environments

Jae-Kwang Lee*, Jongtae Jeong, and Min-Hoon Baik

Korea Atomic Energy Research Institute (KAERI)

*jkleel@kaeri.re.kr

Disposal methods of radioactive waste can be mainly divided into near-surface disposal and deep geological disposal. If the radioactive waste is exposed to groundwater for a long time in the disposal environment, no matter how the decommissioning waste generated from the nuclear power plant is disposed of, the radionuclides may be released from the disposal site. Decommissioning waste from nuclear power plant contains radionuclides that are harmful to ecosystem including humans. Radionuclides released from disposal site behave in a complex and sensitive manner affected by geochemical conditions such as soil, geological media and groundwater. Sorption is one of the important mechanisms to retard the migration of radionuclides in a subsurface environment. In this study, geochemical properties of groundwater were collected and analyzed in the disposal environment considering disposal method in order to evaluate the geochemical behavior of radionuclides. The formation of aqueous and precipitated species of cesium and cobalt in a disposal condition were calculated and estimated. The sorption properties were also evaluated and predicted by considering the changes in the geochemical conditions such as pH, redox potential and geological media for the safety assessment.

Keywords: Cesium, Cobalt, Disposal, Radionuclide, Safety Assessment, Sorption

Establish Regulatory System and Case Study Database for Deep Geological Disposal System

Jeong Hyun Lee¹, Sunggyun Jang¹, Sang Hun Jeon¹, Gun Hee Kim¹, Ji Eun Shin¹, Si On Kim¹,

and Won-Seok Kim^{1,2,*}

¹NCsquare Co.

²Pohang University of Science and Technology (POSTECH)

*wkim23@ncsquare.co.kr

Kori-1, the nuclear power plants in South Korea, first started operation in April 1978 and was suspended permanently in 2017. The saturation rate time of spent nuclear fuel generated by major nuclear power plants operating in Korea are getting closer. If we fail to dispose spent nuclear fuel, which is equivalent to high-level radioactive waste, the nuclear power plants will have to be shutdown. High-level radioactive waste is permanently disposed through a deep geological disposal system because it contains long-term half-life nuclides and emits high energy. To select the deep geological disposal site and construct the disposal facilities, it is necessary to establish appropriate regulatory policies accordingly.

The status of database construction in OECD-NEA, NRC, SITEX, and IAEA, which provides safety regulations for deep geological disposal system, stipulates each requirement for dismantling nuclear power plants. However, details such as specific figures are not specified, and guidelines for the disposal of high-level radioactive wastes are not clearly distinguished.

In Korea, the CYPRUS program, an integrated database system, has been developed to support comprehensive performance evaluation for high-level waste disposal. However, due to several difficult situations, maintenance and upgrades have not been performed, so the research results exist only in the form of raw data and the new research results have not been reflected. Other than that, there is no preemptive basis for regulating the deep geological disposal system.

With real-time database, we can develop a regulatory system for the domestic deep disposal system by systematically analyzing the regulatory condition and regulatory case data of international organizations and foreign leading countries. The database system processed and stored primary data collected from nuclear safety reports and other related data. In addition, we used relational database and designed table to maximize time and space efficiency. It is provided in the form of a web service so that multiple users can easily find the data they want at the same time.

Based on these technologies, this study established a database system by analyzing the legal systems, regulatory standards, and cases of major foreign leading countries such as Sweden, Finland, the United States, and Japan. This database aims to organize data for each safety case component and further prepare a safety regulatory framework for each stage of development of disposal facilities suitable for the domestic environment.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Foundation of Nuclear Safety (KOFONS) grant funded by the Korea government (Nuclear Safety and Security Commission, NSSC) (No. 1075001193).

Keywords: Deep Geological Disposal System, High-level Radioactive Waste, Database, Development of Regulatory Standards

A Study on the Thermal Conductivity of Bentonite-Sand Mixed Backfill According to Size and Ratio of Sand for Deep Geological Disposal

Yohan Cha*, Jae Owan Lee, Mihye Kong, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*yohan@kaeri.re.kr

The backfill refills the deep geological disposal system after the installation of buffer in the disposal hole. SKB and Posiva have established the safety function for the backfill such as hydraulic conductivity of 10^{-10} m/s and swelling pressure of 0.2 MPa. The study on the thermal properties is required for the evaluation of performance design and long-term stability of backfill, since the thermal condition affects the hydraulic and mechanical behavior of backfill. Thermal conductivity is a key characteristic of thermal properties due to heat dissipation from spent fuel. In this study, thermal conductivities of bentonite-sand mixed blocks were measured. The silica sands were used instead of the crushed rock with bentonite-WRK, one of the candidate bentonite of the Korean repository system. The effects of size distribution and mass ratio of sand were evaluated. Four different size of silica sand (i.e., 0.18-0.25, 0.7-1.12, 1.6-2.5, 2.5-5.0 mm) and five mixing ratio (i.e., 1:9, 2:8, 3:7, 4:6, 5:5 of bentonite and sand) were used for characterization of thermal conductivity. As a result, the thermal conductivities were measured ranging from 1.6 to 3.1 W/m·K depending on the size and mass ratio of the sand. The smaller the size or higher the mixing ratio of sand or the higher the water contents, the higher the thermal conductivity on the surface of backfill block. The higher compressing pressure induce higher thermal conductivity. Meanwhile, the feasibility study of backfill block productivity was reviewed according to the variables of this study. The excessive sand ratio and water contents lead to poor quality that results in the failure of the block. In Korea, the research of backfill is only now in fundamental steps, thus the results of this study are expected to use for setup the experimental conditions of hydraulic and mechanical performance, and can be used for the design of safety function and evaluation of long-term stability for deep geological disposal system.

Acknowledgements

This work was supported by the institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (2021M2E1A1085193).

Keywords: Disposal Repository, Backfill, Thermal Conductivity, Performance Requirement

Experimental Study on Piping Erosion in Compacted Bentonite Buffers Considering Buffer-Rock Interactions

Minhyeong Lee*, Chang-Ho Hong, Ji-Won Kim, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*leemh@kaeri.re.kr

When a rapid groundwater inflow is introduced from the adjacent rock mass in the early stage of disposal, hydraulic pressure build-up occurs, which may cause piping erosion at the buffer material itself and the interface of the gap-filling material. Such piping erosion in compacted bentonite buffer via interaction between the buffer and the adjacent rock mass may deteriorate the performance of the buffer material. Therefore, it is necessary to understand the conditions and scenarios in which the piping phenomenon around the buffer material occurs for the long-term health of the repository. In this study, laboratory-scale experimental tests of piping erosion in buffer and interfacial rock was introduced. $\varnothing 100 \text{ mm} \times 200 \text{ mm}$ height compacted bentonite specimens were placed in a cylindrical acetal cell, and the distilled water was continuously injected at a flow rate of 0.068 L/min using a dual syringe pump. The inflow of water was generated from the bottom and side cell of buffer material. During water injection, injected water pressure and amount were measured with visual observation. The results showed that the external saturation of buffer firstly occurs followed by piping crack generation along the wetting front. The additional piping channels were generated and merged with others. As the injection stopped, the swelling and self-sealing behavior of buffer material were observed. Moreover, X-ray CT scanning of the cell was conducted after the piping simulation to analyze the piping channels and saturation depth. The results highlight the piping erosion phenomenon mainly occurs due to the presence of a gap outside the buffer material. Further experimental cases is need to comprehensively understand piping phenomena in buffer material for assessing the long-term stability of underground radioactive waste disposal systems.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (NRF-2020M2C9A1062949 and NRF-2021M2E1A1085193) funded by the Ministry of Science and ICT.

Keywords: Engineered Barrier System, Groundwater Inflow, Piping Erosion, Bentonite Buffer, X-ray CT

Investigation of Granitic Groundwater Colloids in KURT System by Using Various Analytical Methods

Jun-Hyuk Ha and Jun-Yeop Lee*

Pusan National University (PNU)

*jylee@pusan.ac.kr

According to the continued generation of spent nuclear fuel, a reliable safety assessment is highly required with the precise modeling of the migration and retardation behavior of radionuclides to enhance public acceptance and hinder excessive conservativeness during the construction of the repository. In particular, the colloids formed in the repository-relevant condition are known to accelerate the migration of radionuclides. Thus, geochemical behavior and relevant characteristics of colloids are needed to be unambiguously clarified. The objective of the present work is to investigate the fundamental characteristics of colloids contained in the natural groundwater system by using various analytical methods and the tangential flow ultra-filtration (TFUF) system.

The granitic groundwater sample from the DB-3 borehole at the KURT (KAERI Underground Research Tunnel) was taken by an airtight stainless steel cylinder coated on the inside with PTFE to prevent the infiltration of ambient air into the geologic groundwater sample. And then, the groundwater sample was transferred to the inert glovebox filled with Ar gas to monitor the pH and Eh equilibrium of the aqueous sample. For further investigation, the colloid contained in the groundwater sample was concentrated by using the TFUF system equipped with a membrane filter (pore size: 3 kDa). The concentrated groundwater sample was analyzed with various methods such as ICP-MS/OES, IC, DLS/ELS, FE-TEM/SEM-EDS, ATR-FTIR, TOC, LC-OCD, etc.

In this study, the size of groundwater colloids was determined to be 182.3 ± 52.7 nm with the major constituents of C, S, O, Fe, Al, Si, etc. The amount of organic carbon and the concentrations of organic substances determined by means of the molecular weight fraction with the TOC and LC-OCD provide further detailed information for the colloids in the KURT groundwater sample. The results obtained in this study are expected to be used as preliminary experimental data for modeling the colloid-facilitated migration of radionuclides to improve the reliability of the safety assessment of the geologic repository.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (No. NRF-2021M2E1A108520412).

Keywords: Colloid, KURT, Natural Groundwater, TFUF

Experimental Study on Frictional-Healing Behavior of a Rock Joint

Seungbeom Choi, Yongki Lee, and Taehyun Kim*

Korea Atomic Energy Research Institute (KAERI)

*kth@kaeri.re.kr

A rock joint exerts significant influences on the rock mass behavior in terms of thermal, hydraulic, and mechanical (THM) aspects. Therefore, its features should be thoroughly investigated in various rock mechanical projects, such as high-level radioactive waste (HLW) disposal repository, tunnel, and rock slope. Meanwhile, it is essential to guarantee the safety of the disposal repository for a very long period of time and it should prepare measures for various risks, which may possibly encounter during that period. In general, direct shear tests for a rock joint are conducted to investigate the possibility of frictional sliding of the joint under specific loading conditions or to predict the shear strength of the joint. However, it is necessary to consider whether regional sliding of a rock joint or reactivation of a fault might occur due to an earthquake or redistribution of the in-situ stresses because the expected operation period of the repository is quite long, and various situations can happen. A slide-hold-slide test for a rock joint is a practical test that can investigate the time-dependent behavior or frictional-healing of a joint. The test enables an estimation of the stress build-up phenomenon after strain energy release in a quantitative manner. In this study, a series of slide-hold-slide tests were carried out in order to investigate the characteristics. Joint specimens were made from mortar, which is a rock-like and brittle material, so as to consider the effect of joint roughness and to secure the reproducibility of the tests. At the same time, mechanical conditions as well as thermal and hydraulic were applied in order to take the environment of the repository into account. As a result, the behavior of shear stress recovery was observed, and the effects of THM coupled condition on the recovery were investigated. This study presents fundamental results of the experiments, and further research outcomes, including time dependent behavior of a joint, will be presented sequentially.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (2021M2E1A1085193).

Keywords: Rock Joint, Direct Shear Test, Slide-hold-slide Test, THM Coupled Condition

Factors of Geological Risk Assessment for Deep Underground Tunneling in South Korea

Myeong-Hyeok Ihm*

Daejeon University

*mhilm61@hanmail.net

For the geological risk assessment of deep-depth underground condition by excavation work or tunneling, since rocks and geologic structure of each country is different, it is necessary to objectify or classify quantitatively deep-depth underground risk evaluation in accordance with Korean geologic characteristics. It could be summarized major factors of rock failure and underground space deformation by geological and geotechnical features as geologic structures, overburden, rock mass characteristics, groundwater, high stress and additional categories. Induced main factors that could be identified and predicted intermediate to deep-depth underground risk through literature investigation and analysis study on research trend related to the underground geological engineering. In order to assessment the risk of rock mass excavated from 100 m or more to several kilometers deep below the ground are classified into about 19 factors, and can be divided into 6 categories. Using these risk factors as basic data, weights for each factor for each category can be set, and further, the risk of excavated rock mass can be calculated.

Keywords: Risk Factors, Geologic Structure, Rock Type, Rock Mass Characteristics, Groundwater

Evaluation for Compaction Curves of Compacted Bentonite-Sand Mixtures

Deuk-Hwan Lee, Seok Yoon*, Jin-Seop Kim, Ji-Won Kim, and Gi-Jun Lee

Korea Atomic Energy Research Institute (KAERI)

*syoon@kaeri.re.kr

The buffer block, which is one of the main components of the engineering barrier system, plays an essential role in mitigating groundwater infiltration and radionuclide transport in a high-level nuclear waste repository. To achieve those purposes, the compacted buffer block must satisfy the functional safety criteria for dry density, water content, and many other components. In this study, the compaction curves of the compacted bentonite-sand mixtures were evaluated to identify the relationship between the dry density and the water content of the buffer material. The floating die press at 10 MPa and the cold isostatic press at 40 MPa were applied to compaction of a buffer block with a diameter of 100 mm and a thickness of 10 mm. The condition of a bentonite-sand mixing ratio was 6:4, 7:3, 8:2, and 9:1 with 9 to 21% water content. As a result, the maximum dry density increases, the optimum moisture content decreases as the sand content of buffer material increases. This study can provide the conditions for manufacturing the compacted bentonite-sand buffer block.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (2021M2E3A2041351) and the Institute for Korea Spent Nuclear Fuel and National Research Foundation of Korea (2021M2E1A1085193).

Keywords: Bentonite-sand, Compaction Curve, Buffer Material, Deep Disposal Repository

Assessment of Evapotranspiration Around KAERI Underground Research Tunnel

Byeong-Hak Park*, Eunhye Kwon, and Kyung-Woo Park

Korea Atomic Energy Research Institute (KAERI)

*bh-park@kaeri.re.kr

The change of surface environments (e.g., climate change, uplift/subsidence, and erosion) can undermine the long-term safety of a high-level radioactive waste repository. Therefore, understanding the water cycle between atmosphere, surface, and subsurface is essential to ensure the long-term safety of deep geological disposal and consequently to gain public acceptance for the repository. Among hydrologic components (e.g., precipitation, interception, runoff, infiltration, evapotranspiration (ET), and recharge) which constitute the water cycle, ET is more than half of the total precipitation and plays a crucial role in the water and energy transfer among the three systems. Although various methods for ET evaluation (e.g., Bowen Ratio, Eddy Covariance, Optical Scintillation, and Weighing Lysimeter methods) have been developed, many influential factors such as vegetation, climate, and moisture content make its accurate evaluation still tricky. In this work, we chose weighing lysimeter and Penman-Monteith methods for direct/indirect estimation of ET, and installed a smart field lysimeter and a micro-meteorological station around KAERI Underground Research Tunnel. Water balance in the unsaturated zone and five climatic variables (air temperature, humidity, precipitation, radiation, and wind speed/direction) were measured more than once per 10 minutes for six months from April to September, 2022. From the measurements, daily actual and potential ET values at the study site were calculated and compared. We also discussed the applicability and limitation of current methods and ET assessments at different spatial scales regarding verifying and validating the developing numerical models.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korean government (Ministry of Science and ICT, MIST) (No. 2021M2E1A1085200).

Keywords: Safety, Water Cycle, Hydrologic Process, Evapotranspiration, KURT

Microbial Effects on the Corrosion of Metals in Compacted Bentonite Blocks

Seung Yeop Lee*, Jae-Kwang Lee, Sang-Ho Lee, Jin-Seok Kim, and Jang-Soon Kwon

Korea Atomic Energy Research Institute (KAERI)

*seungylee@kaeri.re.kr

Safe storage of spent nuclear fuel in deep underground repositories needs an understanding of the long-term alteration (corrosion) of metal canisters and buffer materials. We conducted a small-scale laboratory alteration tests on some metal (Cu and Fe) chips by embedding them into the compacted bentonite blocks, which were placed in anaerobic water for 1 year. Some additives like lactate, sulfate, and bacteria were separately loaded into the water to promote biochemical reactions. The bentonite blocks immersed in the water were finally dismantled after 1 year, and they showed that their alteration was insignificant. However, the Cu chip exhibited some microscopic etch pits on its surface, wherein sulfur component was slightly detected. Overall, the Fe chip was more corroded than the Cu chip under the same condition. The secondary phase of the Fe chip was locally found as carbonate materials, such as siderite (FeCO_3) and calcite ($(\text{Ca}, \text{Fe})\text{CO}_3$). These secondary products could imply that the local carbonate production around the Fe chip may be initiated by an evolution (alteration) of bentonite and a diffusive provision of biogenic CO_2 gas. These laboratory scale test results suggest that the long-term alteration (corrosion) of metal canister/bentonite blocks in the engineered barrier could be possible and may be promoted by microbial activities.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085202).

Keywords: Repositories, Corrosion, Canisters, Bentonite, Bacteria

Modeling of Uranium Migration Near High-Level Radioactive Waste Repository Using COMSOL-PHREEQC Coupled Model

Won Woo Yonn¹, Weon Shik Han^{1,*}, Jeonghwan Hwang², and Jung-Woo Kim²

¹Yonsei University

²Korea Atomic Energy Research Institute (KAERI)

*hanw@yonsei.ac.kr

The analysis of uranium migration is crucial for the accurate safety assessment of high-level radioactive waste (HLW) repository. Previous studies showed that the migration of the uranium can be affected by various physical and chemical processes, such as groundwater flow, heat transfer, sorption/desorption and, precipitation/dissolution. Therefore, a coupled Thermal-Hydrological-Chemical (THC) model is required to accurately simulate the uranium migration near the HLW repository. In this study, COMSOL-PHREEQC coupled model was used to simulate the uranium migration. In the model, groundwater flow, heat transfer, and non-reactive solute transport were calculated by COMSOL, and geo-chemical reaction was calculated by PHREEQC. Sorption was primarily considered as geo-chemical reaction in the model, using the concept of two-site protolysis nonelectrostatic surface complexation and cation exchange (2 SP NE SC/CE). A modified operator splitting method was used to couple the results of COMSOL and PHREEQC.

Three benchmarks were done to assess the accuracy of the model: 1) 1D transport and cation exchange model, 2) cesium transport in the column experiment done by Steefel et al. (2002), and 3) the batch sorption experiment done by Fernandes et al. (2012), and Bradbury and Baeyens (2009). Three benchmark results showed reliable matching with results from the previous studies. After the validation, uranium 1D transport simulation on arbitrary porewater condition was conducted. From the results, the evolution of the uranium front with sequentially saturating sites was observed. Due to the limitation of operator splitting method, time step effect was observed, which caused the uranium to sorbed at further sites than it should. For further study, 3 main tasks were proposed. First, precipitation/dissolution will be added to the reaction part. Second, multiphase flow will be considered instead of single phase Darcy flow. Last, the effect of redox potential will be considered.

Acknowledgements

This research was supported by the Nuclear Research and Development project through the National Research Foundation of Korea (NRF) funded by Ministry of Science and ICT [project number: 2021M2E1A1085187], and by Basic Science Research Program through the National Research Foundation (NRF) of Korea funded by the Ministry of Education [project number: 2016R1D1A1B01008715].

Keywords: High-Level Radioactive Waste Repository, THC Model, Uranium Migration, Safety Assessment

Development of a Numerical Module to Convert DFN to ECPM

Jung-Woo Kim*, Yong-Min Kim, Jaewon Lee, and Sung-Hoon Ji

Korea Atomic Energy Research Institute (KAERI)

*jw_kim@kaeri.re.kr

In the geological disposal system whose host rock is crystalline rock, fractures play a significant role in the safety assessment as they are the main pathway of the radionuclide migration. From the perspective of long-term safety assessment, the properties of fractures can be changed by tectonic movement such as earthquake, uplift, etc. In general, methods for simulating fractures include Discrete Fracture Network (DFN), which directly simulates the fracture surface, and Equivalent Continuous Porous Media (ECPM), which is equivalent to the ratio of the fractures in a certain rock volume. DFN is generally appropriate for deterministic fractures with large scale and high flow velocity, but ECPM may be more appropriate for small scale and sporadically distributed stochastic fractures because the flow velocity is slow and thus the rock matrix diffusion needs to be considered. In fact, several commercial software, such as FracMan, are already in use to convert DFN to ECPM. However, in order to consider the change in properties of fractures due to tectonic movement in the long-term safety assessment, a model that converts DFN to ECPM needs to be modularized and embedded into the safety assessment model. In this study, therefore, an in-house MATLAB code was developed to convert DFN to ECPM, which can be used as a submodule.

The algorithm of converting from DFN to ECPM basically followed the Oda's method. As the first step of the algorithm, in order to obtain the volume ratio of the fracture in a certain mesh element, the cross-sectional area of the fracture and the mesh element was calculated. Then, porosities of each mesh element were calculated as the volume fraction of fractures passing through the mesh element. Based on the Oda's method, the permeability tensors of each mesh element were calculated by using an empirical fracture tensor which is weighted by the cross-sectional area and transmissivity of each fracture. Finally, the newly developed module was verified by a benchmark test, in which the ECPM results converted from a certain DFN data by using the numerical module developed in this study were compared with those by using FracMan. The newly developed module will be installed in the process-based total system performance assessment framework (APro) being developed by KAERI.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: DFN, ECPM, Modularization, MATLAB Code, APro

An Efficient Thermal Management Method of Decay Heat in a Disposal Container for Deep Geological Disposal

Jongyoul Lee*, Heuijoo Choi, and Dongkeun Cho

Korea Atomic Energy Research Institute (KAERI)

*njylee@kaeri.re.kr

Currently, the most widely accepted disposal concept for long-term isolation of high level radioactive waste including spent nuclear fuels is to disposal in a deep geological repository designed and constructed with multiple barriers composed of engineered and natural barriers so that the waste can be completely isolated in a stable deep geological environment. In this concept, an important consideration is the heat generated from the waste due to the large amount of fission products present in the high level waste loaded in the disposal container. For safe and complete isolation of high level radioactive waste in the deep geology, the disposal concepts that meet the thermal requirements for the disposal system design have been developed by harmonizing the thermal characteristics of engineered and natural barriers in Korea. In this paper, the deposition hole configuration and the decay heat dissipation area (surface area) of disposal container were considered for the efficient thermal management in the deep geological disposal concept. Heat transfer through the waste form, its container and surrounding components and the rock will be mainly by conduction. Heat transfer by radiation and convection can be negligible after backfilling. When considering heat conduction, according to Fourier's law, if the thermal conductivity of the repository components is the same, the greater the heat dissipation area and the adjacent temperature gradient, the greater the conduction effect. Therefore, rather than the conventional concept of loading 4 PWR spent fuel assemblies per disposal container and placing one disposal container in a deposition hole, it is better to load one assembly per disposal container and place 4 disposal containers in a deposition hole. In this case, it was found that the disposal area could be reduced through efficient thermal management. Considering this thermal management method as an alternative to the concept of deep geological disposal, additional research is needed.

Acknowledgements

This work was supported by the Ministry of Science and ICT within the framework of the national long-term nuclear R&D program (NRF-2021M23A2041312).

Keywords: Spent Nuclear Fuels, Disposal Container, Decay Heat, Deep Geological Disposal, Thermal Conductivity, Thermal Management

Acoustic Emission Characteristics for Bending Failure of Engineered Barrier

Jong-Won Lee¹, Hyunwoo Kim², Min-Jun Kim², and Tae-Min Oh^{1,*}

¹Pusan National University (PNU)

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

*geotaemin@pusan.ac.kr

The underground environment has an advantage to minimize the external influences because it is isolated space with surrounded rock medium. Therefore, underground rock has been used recently as the target for a disposal system of spent fuel with high-level radioactive. The disposal system mainly consists of natural barrier (i.e., surrounded rock medium) and engineered barrier (i.e., concrete lining, plug, backfill, canister, and buffer). In particular, the engineered barrier is important for long-term storage because it has to preferentially block the leakage of radioactive nuclide.

Non-destructive technologies (NDT) have been utilized to monitor the state of disposal system for considering the limitation in deep depth conditions such as limited environment for direct damage inspection. Acoustic emission (AE) monitoring technique is an effective method to monitor the damage (crack) magnitude, history (i.e., crack evolution), and location using high-frequency elastic waves. To apply the AE monitoring method in the disposal system, the characteristics of damaged materials should be considered. The concrete lining has multi-failure behavior (i.e., brittle and ductile) resulted from composition as cement and reinforcing steel bar. Therefore, it is important to investigate the AE characteristics according to the failure level of reinforced concrete for damage monitoring of the disposal systems.

In this study, the four-point bending tests were carried out to measure the AE signals from the cracking of reinforced concrete specimens in laboratory. The test specimens were prepared with different strength. After the experiment, the AE characteristics were analyzed using the AE parameters with loading and failure state in the curve of time-stress. This study will be helpful for damage monitoring using AE technique in the field of high-level radioactive disposal system.

Acknowledgements

This research was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea Government (MSIT) (No. NRF-2022R1I1A3065299).

Keywords: Acoustic Emission, Bending Failure, Crack Evolution, Disposal System, Engineered Barrier

Bentonite Swelling and Erosion in an Artificial Fracture

Ji-Won Kim¹, Chang-Ho Hong¹, Minhyeong Lee¹, Song-Hun Chong², and Jin-Seop Kim^{1,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Sunchon National University (SCNU)

*kjs@kaeri.re.kr

Compacted bentonite buffer materials are a key component of the engineered barrier system for high-level radioactive waste disposal. The bentonite buffer is saturated via groundwater flow through the excavation damaged zone in the adjacent rock mass. Bentonite saturation results in bentonite swelling, gelation and intrusion into the nearby rock discontinuities. Groundwater flow can cause bentonite erosion and transportation of bentonite colloids. This bentonite mass loss can negatively impact the long-term integrity of the engineered barrier system. Hence, it is necessary to understand the effects of erosion on the properties of the bentonite buffer.

In this study, a series of artificial fracture erosion experiments are conducted to investigate the erosion characteristics of compacted Ca-bentonite buffer materials for different initial dry density conditions. Compacted bentonite blocks and bentonite pellets were manufactured using the cold isostatic pressing technique and granulation compactor respectively. The specimens were placed in a custom-made transparent artificial fracture cell and the bentonite intrusion characteristics were monitored for two months under free swelling conditions with no groundwater flow. The radial expansion of the bentonite specimens within the artificial fracture was measured using a digital camera. In addition, the swelling pressure, displacement, and saturation were determined using a load cell-piston system, LVDT, and electrical resistivity electrodes respectively. A hydro-mechanical-chemical coupled dynamic bentonite diffusion model was applied to model the bentonite erosion characteristics using COMSOL Multiphysics.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (NRF-2020M2C9A1062949 and NRF-2021M2E1A1085193) funded by the Ministry of Science and ICT.

Keywords: Engineered Barrier System, Bentonite Erosion, Artificial Fracture Test, COMSOL Multiphysics

Development and Field Applications of Deep Borehole Hydraulic Testing System (DHTS) Based on Integrated Type Main Frame

Seongho Bae^{1,*}, Hagsoo Kim¹, Eui-Seob Park², Dae-Sung Cheon², and Ki-Bok Min³

¹GeoGeny Consultants Group Inc.

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

³Seoul National University (SNU)

*shbae007@geogeny.biz

The reliable information on the hydraulic characteristics of rock mass is one of the key site factors for design and construction of deep subsurface structures such as geological radioactive nuclear waste disposal repository, underground energy storage facility, underground research laboratory, etc.

In order to avoid relying on foreign field test technology in future projects, we have independently designed and made integrated type main frame, 120 bar waterproof downhole sonde, and 1,200 m wireline cable winch through a series of R&D activities. They are core apparatuses of the Deep borehole Hydraulic Test System (DHTS). Integration of individual test equipment into a single main frame allows safe and efficient work in the harsh field condition. The DHTS was developed aiming primarily for constant pressure (head) injection test and pulse test in deep impermeable rock mass. The maximum testing depth of the DHTS is about 1,050 m from the surface. Using this system, it is possible to make precise stepwise control of downhole net injection pressure in less than 2.0 kgf/cm² with dual hydraulic volume controller and also to inject and measure the very low flow rate below 0.01 l/min with micro flow rate injection/control module.

Over the past two years, we have successfully completed more than 50 in situ hydraulic tests at 5 deep boreholes located in the Mesozoic granite and sedimentary rock regions in Korea. Among them, the deepest testing depth was more than 920 m. In this paper, the major characteristics of the DHTS are introduced and also some results obtained from the high precision field tests in the deep and low permeable rock mass environment are briefly discussed.

Keywords: Hydraulic Characteristics, Integrated Type Main Frame, DHTS, Constant Pressure Injection Test, Pulse Test, Micro Flow Rate

Revisiting A Counter Rotating Plasma Centrifuge For Separating Nuclear Waste

Jun-Ho Seo*

JeonBuk National University (JBNU)

*jhseo@jbnu.ac.kr

Filtering nuclides in high-level nuclear waste using rotating plasmas is a physical separation method based on mass difference. Since it is not chemical separation or metallurgical separation, the elements are separated regardless of their chemical composition. Accordingly, the more the number of chemical elements present in the waste or the more difficult when using the differences in solubility, the more advantageous. However, to be economically competitive, new concepts for rotating plasmas are needed to improve the poor separation rates despite high energy and installation costs. In this work, we revisited a counter rotating plasma centrifuge to examine its potential as an effective device for separating nuclear waste efficiently and economically.

Keywords: Plasma Centrifuge, Separation, Nuclear Waste, High-level

Comparative Mapping of NEA International and Project-Specific FEP Lists for Deep Geological Disposal of Spent Nuclear Fuel

Jaehyeon Yang, Haeram Jeong, Kibeom Son, Donghoon Baek, Karyoung Choi, Gyunyoung Heo,

Kunok Chang, and Hyungdae Kim*

Kyung Hee University

*hdkims@khu.ac.kr

Many countries have been developing their own FEP (Feature, Event, Process) lists to formulate radionuclide release scenarios in deep disposal repository of spent nuclear fuels and to assess the safety. The main issue in developing a FEP list is to ensure its completeness and comprehensiveness in examining all plausible scenarios of radionuclide release in a repository of interest. To this end, the NEA International FEP (IFEP) list as a generic reference have been developed and updated through long-term international collaborations. Leading countries advanced in the research field of deep geologic disposal of spent nuclear fuels have comparatively mapped their project-specific FEP (PFEP) lists with the IFEP list. Recently in 2019, NEA has published an updated version of IFEP list (ver. 3.0) which has a different classification system: the IFEP version 3.0 has the five main categories including the waste package, repository, geosphere, biosphere and external factors while the previous IFEP versions were mainly classified into the external, environmental, and contaminant factors. Most leading countries in this field, Finland and Sweden, recently succeeded to obtain the design and/or construction licenses for deep geologic disposal of spent nuclear fuel. Therefore, their PFEP lists should be good benchmark cases to the following countries. However, their PFEP lists have not comparatively mapped with the most recent version of IFEP and thus some gaps may exist in showing completeness and comprehensiveness in comparison to the IFEP version 3.0. In this study, we comparatively map the PFEP lists of Finland and Sweden to the IFEP version 3.0. The comparatively mapped PFEP list could be used as the basis for verifying the comprehensiveness and completeness of the domestic PFEP list currently under development in Korea.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (Ministry of Trade, Industry and Energy (MOTIE)) (No. 2021040101003C).

Keywords: Spent Nuclear Fuel, Deep Geological Disposal, Safety Case, FEP, Scenario, Comparative Mapping

Conceptual Operation System Design and Procedure for Deep Borehole Disposal Using the Wireline Emplacement Method

Hyun Woo Song^{1,*}, Moonoh Kim¹, Dong-Keun Cho², and Changsoo Lee²

¹KONES Corp.

²Korea Atomic Energy Research Institute (KAERI)

*shw@kones21.com

Despite the increasing interest in Deep Borehole Disposal (DBD) for its capability of minimizing disposal area, detailed research about DBD operation system design should be conducted before the DBD can be implemented. Recently, DBD operation system applying wireline emplacement (WE) technique is under study due to its high flexibility and capability of minimizing surface equipment. In this study, a conceptual WE system, and operation procedure is introduced.

The conceptual WE system consists of 3 main stations, which from the top are hoisting station (HS), canister connection station (CCS) and basement (BS).

In HS, WE is controlled and monitored. The WE is controlled using wireline drum winch and sheaves, and load on wireline is measured using a load cell. HS also has a pressure control system (PCS), which monitors internal pressure of the system, and a lubricator, which act as housing for joint device, allowing the joint device to be easily inserted into the borehole. The joint device is used to connect the disposal canister to wireline for emplacement/retrieval.

In CCS, a rail transporter brings a transport cask containing disposal canisters, then the transport cask is connected to the hoisting system and a PCS in the BS. The main component located at canister station are a sliding shielding door (SSD), and a slip. The SSD is used to prevent canister from falling into borehole during the connecting operation and prevent radiation from BS to affect the workers. The slip is located beneath the SSD and is used to hold the disposal canister before it is lowered into the borehole.

In BS, PCS is installed to prevent overflow and blowout of borehole fluid. The PCS consists of wireline pressure valve, christmas tree and BOP, which all are a type of pressure valve to seal the borehole and release pressure inside the borehole.

The WE procedure starts with transporting transport cask to CCS. The transport cask is connected to lubricator, and PCS. Joint device is lowered down to be connected with disposal canisters, then pulled up to check the load on the wireline. After the check-up, SSD is opened, and disposal canister is lowered into the borehole. When desired depth is reached, joint device is disconnected and retrieved for next emplacement.

In this study, the conceptual deep borehole disposal system design implementing WE technique is introduced. Based on this study, further detailed design could be derived in future, and feasibility could be tested.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) and granted fund by Korea government (MSIT) (No. 2021M2E3A2041312).

Keywords: Deep Borehole Disposal (DBD), Wireline Emplacement, Conceptual Operation System, Operation Procedure

Development of Copper Alloys for Deep Geological Disposal Canisters of High-level Radioactive Waste

Minkyu Ahn¹, Jinwoo Park¹, Sangeun Kim¹, Chansun Shin^{1,*}, and Dongkeun Cho²

¹Myongji University

²Korea Atomic Energy Research Institute (KAERI)

*c.shin@mju.ac.kr

Copper is used for deep geological disposal canisters of spent nuclear fuels, because of excellent corrosion resistance in an oxygen-free environment. However, sulfide formation during the long-term exposure under deep geological disposal condition can be harmful for the integrity of copper canisters. Sulfur around the canisters can diffuse along grain boundaries of copper, causing grain boundary embrittlement by the formation of copper sulfides at the grain boundaries. The development of copper alloys preventing the formation of copper sulfides along grain boundaries is essential for the long-term safety of copper canisters.

In this research, the mechanisms of copper sulfide formation at the grain boundary are identified, and possible alloying elements to prevent the copper sulfide formation are searched through the first principle calculations of solute atom-vacancy binding energy and the molecular dynamics calculation of grain boundary segregation energy. The comparison with the experimental literature results on the mitigation of copper embrittlement confirmed that the theoretically identified mechanisms of copper sulfide formation and the selected alloy elements are valid. Thereafter, binary copper alloys were prepared by using a vacuum arc melting furnace. Sulfur was added during casting of the copper alloys to induce the sulfide formation. The cast alloys were cold-rolled into a plate after homogenization heat treatment. The microstructure and mechanical property of each alloy were investigated after recrystallization in a vacuum tube heat treatment furnace. The copper alloys developed in this study are expected to contribute in increasing the long-term safety of deep geological disposal copper canisters by reducing the embrittlement caused by the sulfide formation.

Keywords: Deep Geological Disposal Canister, Copper Alloys, Embrittlement, Copper Sulfide Formation

Implementation of Specific Activity Models for C-14 in Terrestrial Environment of a Disposal Facility Using GoldSim

Dongki Kim^{1,*}, Joowan Park¹, Seo-Yeon Cho², Mi-Seon Jeong²

¹RADCORE

²Korea Radioactive Waste Agency (KORAD)

*dongki1997@radcore.co.kr

In biosphere assessment modeling for the safety assessment of the Wolsong LILW disposal facility, the multi-compartment modeling in which all radionuclides transport is described quantitatively in terms of transfer factors between various environmental compartments has been implemented. In order to reflect the actual transfer mechanisms of ^{14}C in the environment the specific activity (SA) modeling approach can be applied as an alternative to the previous transfer factors (TF) approach. The assumption of full SA equilibrium throughout the terrestrial environment is completely satisfactory for ^{14}C release to the atmosphere if the ^{12}C is emitted as $^{14}\text{CO}_2$. This is the only form that is readily taken up by plants, so that active carbon is incorporated into the plant via photosynthesis at the same rate as stable carbon. Accordingly, the ^{14}C concentration in Bq/g stable carbon is the same in the plant as it is in the air. And animals take up carbon almost entirely through ingestion and the SA ratio in the plant is maintained in the animal.

In this study, a specific activity model for ^{14}C was implemented in a GoldSim biosphere assessment model. From the literature survey for existing specific activity models developed, the IAEA model was selected. The farming scenario utilizing well water was simulated and the resulting ingestion dose conversion factors (DCFs) from the IAEA SA model were compared with those of the TF approach. The parameter value for the concentration of stable carbon in the air (gC/m^3) is used as $0.20 \text{ gC}/\text{m}^3$ considering the Suess effect. The dose coefficient for food ingestion used for dose calculations was taken from ICRP-72 as $5.8\text{E}-10 \text{ Sv}/\text{Bq}$.

It was found that the ingestion DCFs of the SA model showed about 3 times lower than those of the TF model in the farming scenario through irrigation of well water, so it is expected that the SA approach could be applied for a more realistic assessment. Though the comparisons were made on the results from the terrestrial ecosystem only in this study, it would be necessary to investigate the applicability of the SA modeling approach for ^{14}C through extensive comparisons and analysis including an aquatic ecosystem, and through parameters survey suitable to the domestic condition.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (Ministry of Trade, Industry and Energy (MOTIE)) (No. 2021040101003C).

Keywords: Specific Activity Model, Carbon-14, GoldSim, Biosphere Assessment Modeling

A Study on the Technology Level of High-Level Radioactive Waste Management: Focused on Delphi and Focus Group Interview Survey Method

Hyeon Woo Oh^{1,*}, Jin Beak Park¹, Junghwa Yang², Rinah Kim¹, and Minkyu KIM¹

¹Korea Radioactive Waste Agency (KORAD)

²Korea Mine Rehabilitation and Mineral Resources Corp. (KOMIR)

*nucleoh@korad.or.kr

The 2-round Delphi survey and Focus Group Interview (FGI) survey method, in this study, are sequentially applied for the level analysis of the high-level radioactive waste (HLW) management technologies, that are classified into transport/storage, site evaluation, and disposal categories. The 2-round Delphi survey was conducted on domestic 56 experts in the HLW field in Korea, and survey answers were managed with questionnaires distributed by e-mail. In the FGI survey, domestic 24 experts from management field were formed into three groups to conduct in-depth interviews. Past research achievements including journal papers, intellectual properties and the expert opinions presented at expert hearing on HLW technology were used as reference materials. As a result of the survey, in this study, the average domestic technology level compared to the leading countries was 83.1% in transport area, 79.6% in storage area, 62.2% in site evaluation area, and 57.4% in disposal area, respectively. When compared to the former level analysis results in 2017, technology level of transport-storage area increased by 8.6%, and the site evaluation-disposal technology area decreased by 7.27%. The highest factor that increase the level of technology in the transport-storage field was due to the increased R&D program resulting on journal papers, intellectual properties. In addition, the decrease factor in the level of technology in the site evaluation-disposal field was mainly due to relatively low R&D program when compared to the leading countries. Suggested method for the level survey can be used to find out the basic data of the lower tech technologies, to estimate the efficient research budgets and to prepare the R&D human resources. With this regards, R&D roadmap can be matured with suggested prediction method for the domestic technology level on HLW.

Keywords: High-Level Radioactive Waste, Technology Level, Delphi, Focus Group Interview (FGI)

Evaluation of KJ-II Bentonite Properties Reacted With Highly Concentrated K^+ (1 M) Solutions With Different pHs at 150°C

Ji Hoon Lee¹, Ho Young Jo^{1,*}, and Jang Soon Kwon²

¹Korea University

²Korea Atomic Energy Research Institute (KAERI)

*hyjo@korea.ac.kr

Bentonite has been considered as a buffer material in a deep geological repository for high-level radioactive waste (HLW). Bentonite may come into contact with various chemical solutions during the long-term storage. In particular, solutions containing K^+ can affect the stability of bentonite (e.g., illitization). The bentonite can be gradually saturated with the inflow of groundwater, and the temperature can rise simultaneously due to the decay of HLW. This study aimed to evaluate the bentonite stability in contact with very highly concentrated K^+ solutions with different pHs at 150°C. Batch reaction tests using KJ-II bentonite were performed for 30–150 days in teflon-stainless steel reactors. De-ionized (DI) water (pH = 6.0) and 1 M KCl (pH = 6.0), and 1 M KOH (pH = 12.5) solutions were used as reaction solutions. After completing batch reaction tests, the reacted samples were analyzed using various analytical techniques. For DI water, chemical, mineralogical, and physicochemical properties of reacted samples were similar to those of unreacted samples. For 1 M KCl solutions, cation exchange for Ca by K and slight changes in mineralogical properties of reacted samples were observed, but there are no significant changes in the physicochemical properties. In contrast, for 1 M KOH solutions, changes in chemical, mineralogical, and physicochemical properties of reacted samples were observed. Results of X-ray diffraction (XRD) analysis indicated dissolution of montmorillonite and formation of zeolite minerals, which were confirmed by thermogravimetric-differential thermal analysis (TGA-DTA) and Fourier transform infrared (FTIR) analysis. These results suggest that highly concentrated K^+ (1 M) solution combined with high pH (12.5) and high temperature (150°C) may affect bentonite alteration. These preliminary experiments were intended to qualitatively evaluate the mechanism and influencing factors of the buffer material alteration under extreme experimental conditions, and it is revealed that the conditions do not reflect the actual repository environment.

Keywords: Bentonite, HLW, Illitization, Potassium Ion

Physical and Chemical Properties of Steam-Treated Bentonite

Hong Jae Kim* and Ho Young Jo

Korea University

*kinghong79@korea.ac.kr

Bentonite containing >50wt% montmorillonite is being considered as a buffer material in a deep geological repository to dispose of high-level radioactive wastes (HLRW). Bentonite is considered a buffer material because of its exceptional properties such as high swelling capacity, low hydraulic conductivity, and high radionuclide sorption capacity. The bentonite buffer can be exposed to heat from the radioactive decay of HLRW and to groundwater. Water in bentonite buffer can be converted to steam under elevated temperature and pressure conditions. Previous studies reported contrasting results showing that steam treatment could decrease the swelling capacity due to changes in the surface properties from hydrophilic to hydrophobic or could not change. The contrasting results were probably because different studies used different experimental conditions and methods. Therefore, the effect of steam treatment on the bentonite properties is still unclear. The purpose of this study is to determine how the bentonite properties change after steam treatment, in particular swelling and hydrophilic properties. Two types of bentonite were used for steam treatment and analysis; Gyeongju Ca-bentonite (KJ- II) and Wyoming Na-bentonite (GCL-B). Steam treatment was performed at 150°C in an oven for various periods (7, 30, 60, and 90 days). Free swell test, X-ray fluorescence (XRF) analysis, surface-area measurement (BET), thermal gravimetric analysis (TGA), cation exchange capacity (CEC), and water uptake test were performed on steam-treated bentonite for various periods and raw bentonite. After steam treatment, some properties of steam-treated bentonite changed when compared to raw bentonite. Free swell index, which means the swelling capacity, decreased significantly as the results of previous studies. CEC and BET surface area values depended on the bentonite type. For Wyoming Na-bentonite, in which the dominant interlayer cation is a monovalent cation, CEC and BET surface area values were increased. On the other hand, Gyeongju Ca-bentonite, in which the dominant interlayer cation is a divalent cation, has no change in the above two properties. Results of XRF analysis, TGA, and water uptake test showed that these properties of both bentonites did not change after steam treatment. The results of this study confirmed that steam treatment affected the swelling and physicochemical properties of bentonite, in particular Na-bentonite. Further studies will focus on the surface properties of bentonite to investigate whether the surface properties have changed from hydrophilicity to hydrophobicity, or whether the montmorillonite structure has changed.

Keywords: Bentonite, Buffer, High-level Radioactive Wastes, Deep Geological Repository, Steam Treatment

Modeling the Effects of Excavation Damaged Zone on the Radionuclide Transport

Jaewon Lee* and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*jwl@kaeri.re.kr

Excavation Damaged Zone (EDZ) is created by the excavation of deposition holes and disposal tunnels at high-level radioactive waste repository that causes macro- and micro-fracturing in the surrounding rock. Since EDZ can significantly increase the hydraulic transmissivity in the rock and act as a major pathway of leaked radionuclides, consideration of EDZ in terms of safety assessment is very important. Moreover, long-term stress changes such as stress redistribution due to excavation of nearby deposition holes and disposal tunnels, thermal stress due to temperature rise, effective stress change due to pore pressure change, and swelling pressure of bentonite buffer can increase EDZ size and change in thermal-hydraulic-mechanical properties, and consequently, it can affect the transport of radionuclides. Therefore, in order to analyze the effect of long-term evolution of EDZ on radionuclide transport, it is essential to conduct numerical analysis considering the coupled Thermal-Hydraulic-Mechanical (THM) behavior in EDZ.

In order to simulate the behavior of EDZ, coupled THM model was developed using the Adaptive Process-based total system performance assessment framework for a geological disposal system (APro) proposed by the Korea Atomic Energy Research Institute (KAERI). The concept of damage was introduced to demonstrate the jointed rock as a continuous medium. Among several damage models, Mazars damage model was applied in this study. Mazars damage model is the most well-known model for concrete which has similar behavior with rock as brittle material, and the input data of the model can be easily obtained through laboratory testing. If damage occurs due to the influence of thermal-hydraulic-mechanical coupled behavior at the bedrock, the properties change according to the degree of damage, and as a result, the migration of the radionuclide is affected. Based on this conceptual model, radionuclide transport model in the near field considering the long-term evolution of EDZ was developed. To investigate the effect of EDZ in terms of process-based performance assessment, the modeling results with and without EDZ were compared. Finally, by simulating the coupled THM behavior of EDZ with damage model, the effect of long-term evolution of EDZ on radionuclide transport was investigated.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: High-level Radioactive Waste Repository, Process-based Total System Performance Assessment, APro, Excavation Damaged Zone, Damage Model

Mineralogical and Physicochemical Characteristics of Bentonite in the Naah Bentonite Mine, Gyeongju: Natural Analogue Study of the Bentonite Barrier

Yujin Byun^{1,*}, Chanyoung Seo¹, Ho Young Jo¹, Young Jae Lee¹, Ji-Hun Ryu²,

Hye-Yoon Chong³, and Seong-Wan Park³

¹Korea University

²Korea Atomic Energy Research Institute (KAERI)

³Clariant (Korea) LTD.

*Yujin1014@korea.ac.kr

Deep geological disposal is generally accepted to be the most practical approach to handling radioactive wastes. Bentonite has been considered as a buffer material in deep geological disposal repositories (DGR) for high-level radioactive wastes. Evaluating the effect of short-term bentonite alteration on EBS performance has limitations in safety assessment over thousands of years. Information on bentonite characteristics under various conditions obtained from natural systems can be used to evaluate long-term safety of bentonite buffer. The purpose of this study was to investigate mineralogical and physicochemical characteristics of bentonite in the Naah mine located in Yangnam-myeon, Gyeongju-si for a natural analogue of the bentonite barrier in DGR. A total of 15 samples were collected at regular intervals from the bentonite layer and andesitic lapilli tuff (i.e., parent rock) at the boundary with the bentonite layer. The bentonite layer is located at a depth of about 1 m below the ground surface. Each sample was separated into particles $< 75\ \mu\text{m}$ and particles $< 2\ \mu\text{m}$ through grinding and sedimentation processes. The separated subsamples were characterized mineralogically and physiochemically using various analytic techniques. Bentonite samples have a similar $\text{SiO}_2/\text{Al}_2\text{O}_3$ ratio to the parent rock and a lower $(\text{Na}+\text{K})/\text{Si}$ ratio than the parent rock, indicating depletion of alkali components during bentonitization. The parent rock and bentonite samples have similar mineral composition (i.e., quartz, feldspars, opal-cristobalite-tridymite and montmorillonite). Results of XRD analysis on the randomly distributed particles $< 2\ \mu\text{m}$ indicate that bentonite is mostly composed of Ca-montmorillonite, which is a typical dioctahedral smectite. Results of FTIR and VNIR analysis indicate that montmorillonite contained in bentonite is Al-dioctahedral montmorillonite, and Al is substituted with Mg in some octahedron units. The mineralogical and physicochemical characteristics are similar regardless of sampling location. These results suggest that bentonite potentially exposed to weathering, located near the ground surface, has hardly altered.

Keywords: Bentonite, Buffer, Natural Analogue, Deep Geological Disposal Repository

Potential Backfill Materials for Deposition Tunnels in a High-Level Waste Repository

Jae Owan Lee*, Yohan Cha, and Jin-Seop Kim

Korea Atomic Energy Research Institute (KAERI)

*jolee@kaeri.re.kr

Backfill is one of the main components of engineered barrier in a high-level waste repository. The material selection of the backfill determines the barrier performance of the backfill. Overseas, its related research has been carried out mainly in Sweden, Finland, Canada, and Japan. However, Korea has recently started backfill research, and it is urgent to select a potential material for establishing the concept of backfill material and conducting backfill research. This study reviews NEA report, potential materials for overseas backfill research, advantages and disadvantages of single and mixed backfill materials, cases of license applications in Finland and Sweden for the selection of potential materials for backfill in Korea's high-level waste repository.

The review results indicated that it is reasonable to carry out backfill research according to the following plan: Both single and mixed materials are considered as potential materials for backfill research; experiments and performance studies are conducted with these materials; and, based on the results, a potential material or candidate material for the backfill suitable for the HLW repository in Korea is determined. For this plan, the single material is tentatively selected, as in Sweden, as bentonite with a montmorillonite content of about 40-50%. Then, if the selection criteria for montmorillonite content are determined through experiments and performance studies, we determine the final potential backfill material. As for the mixed backfill material, the bentonite/crushed rock mixture seems to be more advantageous than the bentonite/sand mixture considering the disposing problem of crushed rock generated from tunnel excavation and economic feasibility through its recycling. It is thought that the bentonite used in the bentonite/crushed rock mixture should have a higher montmorillonite content than bentonite used as a single backfill material since the crushed rock acts as an inert material in the mixture.

The results of this study can be used as basic data for selecting the backfill material to be applied to the high-level waste repository in Korea, and can be used as a guideline for selecting the potential material required for backfill experiments and performance studies to be carried out in the future.

Acknowledgements

This work was supported by the institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (2021M2E1A1085193).

Keywords: High-level Waste Repository, Backfill, Safety Functions and Requirements, Potential Materials, Candidate Materials

Montmorillonite Alteration in Alkaline and Saline Solutions and Its Influences on Retention of Cesium and Iodide by Bentonite Buffer

Ja-Young Goo^{1,2}, Bong-Ju Kim¹, Ho Young Jo², and Jang-Soon Kwon^{1,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Korea University

*jskwon@kaeri.re.kr

Bentonite, which mostly consists of montmorillonite, is considered as a suitable buffer material for disposal of high-level radioactive wastes in deep geological repository due to its high swelling capacity, low permeability, and strong retention capacity of radionuclide migration. Alkaline and saline solutions originated from degradation of cementitious material and seawater intrusion, respectively, may cause the changes in mineralogical and chemical properties of montmorillonite with various processes such as cation exchange within the interlayer, dissolution of montmorillonite, and precipitation of second minerals. In this study, montmorillonite alteration under alkaline and saline environments and its influences on retention of cesium and iodide by bentonite buffer were investigated. The reactions of bentonite (Bentonil-WRK) with alkaline solutions (0.1 M KOH and NaOH) and simulated saline solution were performed for 7 days in batch experiments at 25°C. After the experiments, reacted bentonite samples were characterized by X-ray diffraction (XRD), Fourier Transform Infrared (FTIR) spectroscopy, Short Wavelength Infrared (SWIR) spectrometry. The concentrations of cesium and iodide dissolved in the solutions were analyzed using an inductively coupled plasma mass spectrometer (ICP-MS). The XRD patterns showed significant decrease in the interlayer space of montmorillonite after the reaction with alkaline solution due to cation exchange and change in hydration status at the interlayer. The retention of cesium and iodide in alkaline and saline solutions were affected by montmorillonite alteration and ion competition. Therefore, the montmorillonite alteration affecting the nuclide retention capacity and long-term stability of bentonite buffer should be considered in the safety assessment of long-term geological disposal performance.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085202).

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

How to Figure out Aperture Distribution in a Fractured Rock

Chung-Kyun Park*, Bong-Ju Kim, Ki-Jong Jang, and Jang-Soon Kwon

Korea Atomic Energy Research Institute (KAERI)

*ckpark@kaeri.re.kr

When the radioactive nuclides are leaked from a deep geological repository by groundwater, the migration path of the nuclides is mostly consisted of rock fractures to the surface biosphere. Thus, assessing the safety of the disposed radioactive wastes depends upon understanding of nuclide migration in the fractured rocks. Fractures in rocks tend to dominate the hydrological characteristics of the dissolved nuclides. To study migration of nuclides in the rock fracture, a granite block of 1 m scale was quarried from the Hwangdeung site. The block has a single natural fracture. The six faces of the rock including fracture gaps were sealed with silicone adhesives to prevent leaking or diffusion of the water.

Usually flow in fractured rock is unevenly distributed and most of the water flow occurs over a small portion of the fracture zone, that is so called channeling flow. It is caused by uneven distribution of apertures in a fracture field. Flow rate is proportional to the cubic of the aperture. Thus, figuring out aperture distribution in a fracture field is the most important step on the study of the migration of nuclides in the fractured region.

The ideal way to figure out the aperture distribution in a fractured rock is to use a non-destructive tool such as X-ray tomography. However, it has a limitation of scale, that is, less than about 30 cm. It is not easy to give a good resolution for this quarried rock of 100×60×60 cm scale. It gives complex and vague images of the fracture. The optimum way to get an aperture distribution in a fractured rock is to drill some boreholes to the fracture and to carry out hydraulic tests. The more number of boreholes gives the more accurate information, but the more disturbance to the fracture field. Thus, it is necessary to optimize between aperture information and disturbing fracture field by selecting a suitable number of boreholes.

We drilled nine boreholes from the upper surface of the rock mass just to the fracture without penetrating the fracture. And we carried out dipole tests for the matrix set of 9 boreholes. From each dipole test, an effective average aperture was calculated with the data of flow rate and hydraulic head. Then aperture distribution in the fracture field is calculated with a modified Krigging method. As a result, the aperture is distributed in the range of about 0.03~0.16 mm.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085202).

Keywords: Fracture, Migration, Aperture, Channeling Flow, Dipole Test

Source Term Evaluation Using ORIGEN-S and TRITON for Spent Nuclear Fuel Disposal

Seo-Yeon Cho*, Hyungju Yun, and Mi-Seon Jeong

Korea Radioactive Waste Agency (KORAD)

*sunnycho@korad.or.kr

In order to dispose of spent nuclear fuel (SNF) in deep geological repository, source term evaluation considering its specification, enrichment, burnup, cooling time should be performed. In this study, the measured values of Takahama-3 pressurized water reactor SNF (WH 17×17) samples were analyzed with SCALE 6.1/ORIGEN-S and TRITON code calculation results for validation. Unlike the ORIGEN-S code, TRITON code calculations differed from two-dimensional neutron flux distribution by using the multi-group cross-section library. Both calculation results from ORIGEN-S and TRITON code showed higher errors in ^{234}U , ^{239}Pu , and ^{241}Pu compared to other actinide nuclides. In the case of axial locations of fuel rods in fuel assembly, fuel rods located at the edge of the fuel assembly presented increased errors due to nuclear reaction cross-section. Overall, the ORIGEN-S predictions informed more accurate agreement with the measured results compared with TRITON results. Especially to ^{235}U , ^{239}Pu , and ^{240}Pu radionuclides, ORIGEN-S errors were denoted more than twice as low as the TRITON results. Comparing the calculation results with experimental results implied that the ORIGEN-S code was more accurate code than the TRITON code for source term evaluation.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government Ministry of Trade, Industry and Energy (MOTIE) (No. 2021040101003C).

Keywords: Source Term Evaluation, Spent Nuclear Fuel, Deep Geological Repository, ORIGEN-S, TRITON

Evaluation of Structural Integrity of Disposal Container Emplacement and Retrieval Device Under Lifting Condition

Young-Oh Lee¹, Moon-Oh Kim^{1,*}, Dong-Keun Cho², and Chang-Soo Lee²

¹KONES Corp.

²Korea Atomic Energy Research Institute (KAERI)

*mokim@kones21.com

The Deep Borehole Disposal (DBD) method has various advantages, such as minimizing the use of site area and corrosion of the disposal container and improving long-term structural safety. However, it is necessary to review the problems that may occur in various technologies related to the emplacement and retrieval of the disposal container and the sealing of the borehole. Therefore, the purpose of this study is to evaluate the structural integrity of an emplacement and retrieval device (hereinafter, the disposal container connecting device) of a DBD container. The disposal connecting device was evaluated according to ANSI 14.6 and NUREG-0612 standards. The allowable stress should be less than the yield strength under the load condition of 3g. The length of the disposal container connecting device was about 2,900 mm, the diameter was 406 mm, and the weight was about 1.2 tons. In addition, 10 disposal containers weighing up to 2.2 tons were handled. The disposal container connecting device was made of stainless steel, and the maximum operating temperature was about 300°C. For structural evaluation, ABAQUS finite element analysis program was used. The analysis model was modeled only 1/2 part considering symmetry condition. The analysis model was modeled using 410,431 nodes and 344,119 solid elements. Three times load was applied to the weight of the disposal container. Axisymmetric conditions were applied to the symmetrical surface of the disposal container, and vertical restraints were applied to the upper lifting lugs. A surface-to-surface contact condition was applied to the part where the contact occurred. As a result of the analysis, the greatest stress was generated at the part supported by the clamp at the disposal container connector at 168.9 MPa. In the lugs and pins connecting the guide and the connecting device, a stress of 530.1 MPa was generated by shearing. In the bolts of the disposal container connecting device, a stress of 498 MPa was generated and the safety margin was 1.73. A stress of 486.1 MPa was generated in the disposal container connecting device, and the safety margin was the smallest 1.16. As a result of the analysis, all components of the disposal container connecting device showed a safety margin of 1.16 or more at the maximum operating temperature and satisfied the allowable stress.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) and granted fund by Korea government (MSIT) (No. 2021M2E3A2041312).

Keywords: Structural Integrity, Deep Borehole Disposal Container Emplacement/Retrieval Device, Lifting Conditions, Deep Borehole Disposal (DBD) Method

Failure Mode of Piezoelectric Sensor for Accelerated Life Test

Changhee Park¹, Hyun-Joong Hwang¹, Gyecheon Cho^{1,*}, and Chang-Ho Hong²

¹Korea Advanced Institute of Science and Technology (KAIST)

²Korea Atomic Energy Research Institute (KAERI)

*gyechun@kaist.ac.kr

The high-level nuclear waste (HLW) repository is a 500-1,000 m deep geological disposal system with a very long life expectancy for disposing of high-level waste, which is known to have a half-life of several thousand years. This repository is subject to harsh environmental conditions, such as high temperature and radiation from high-level waste, that can cause deterioration and crack. When radiation escapes through cracks, it can injure persons on the ground. Therefore, it is essential to install a sensor that can detect problems such as cracks. But, since the high-level nuclear waste (HLW) repository is sealed with bentonite and backfill, the sensor cannot be removed or replaced once it has been installed. Therefore, it is necessary to develop a highly durable monitoring sensor that can withstand harsh environmental conditions. Before attempting to improve durability, it is first required to assess durability quantitatively. And an accelerated life test is a widely used method for assessing durability. However, it is important to obtain the same failure mode when conducting a reliability test, such as an accelerated life test. If the accelerated life test is conducted using different failure modes, the dependability of the results is inevitably diminished. Therefore, in this study, a representative failure mode for the piezoelectric sensor used in the accelerated life test was derived through experiments and literature research.

Acknowledgements

This research was supported by the Nuclear Research and Development Program of the National Research Foundation of Korea (2022M2E3A3015608) funded by the Minister of Science and ICT.

Keywords: The High-level Nuclear Waste (HLW) Repository, Monitoring Sensor, Accelerated Life Test, Failure Mode

Low-pH Concrete Design for Deep Geological Repository

Seok-Gu Gang* and Jongwon Jung

Chungbuk National University

*seokgugang@chungbuk.ac.kr

Spent nuclear fuel still emits radionuclides and high heat that are dangerous to humans. In order to permanently isolate such spent nuclear fuel from human living areas, research is underway to construct a deep disposal system (500 m underground bedrock) consisting of natural and engineering barriers. In this study, plugs, which are engineering barriers consisting of disposal containers, buffer, backfill and plugs were investigated. The plug is one of the engineered barriers made of concrete to block the outflow of radioactive materials and the ingress of organisms, through the tunnel cross-section seals that are eventually discarded. General concrete leachate has a pH of 12.5 or higher and is highly alkaline, which induces dissolution of SiO_2 components contained in the buffer and backfill. Dissolved SiO_2 causes precipitation and cementation on the surface of the buffer and backfill, reducing performance. Therefore, the use of low-ph concrete is essential for deep, high-level waste disposal sites.

Currently, Finland, Sweden, France, Switzerland, etc. have proposed low-ph concrete mix design and performance standards. For example, in Finland, cement, silica fume and fly ash are used as binders and the compressive strength is 50 MPa or more, and the leachate pH is 11 or less. In this research, test specimen fabrication and physical property tests (strength, pH) were performed based on mix design, proposed in Finland, Sweden, France and Switzerland.

A cubic (50 mm×50 mm×50 mm) and a cylinder (Ø100 mm×200 mm) specimens were fabricated. Cubic and cylinder were made of mortar and concrete, respectively, depending on whether they included coarse aggregate.

General concrete strength shows the characteristic that 70 to 80% of the 28th day of the second order appears on the 14th day of the second order and converges after the 28th day. As a result of mortar strength property evaluation, it increased by 30% from 90th to 28th. pH characterization was evaluated according to the powder dissolution method (ESL method) and leaching method (Leachate, EPA 1315) on cubic (mortar) and cylindrical (concrete) specimens, respectively. Mortar pH was measured at 9.78, a decrease of up to 20% from 90 days to 7 days. The physical property evaluation of concrete is currently underway and shows a trend of increasing strength and decreasing pH according to age. Consequently, we aim to present a low-ph concrete mix design for domestic high-level radioactive waste disposal sites.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (2022R1A4A3029737).

Keywords: Deep Geological Repository, Concrete Plugging, Low-ph Concrete, Compressive Strength

Design of Data Structure and User Interface of Data Management Platform for FEP and Scenario

In-Young Kim*, Jongtae Jeong, and Jung-Woo Kim

Korea Atomic Energy Research Institute (KAERI)

*iykim@kaeri.re.kr

Identifying plausible scenarios is necessary to evaluate the performance of the repository reliably over a very long period. All features, events, and processes (FEPs) expected in the repository should be comprehensively well-defined and structured into scenarios based on the relation analysis. A platform for the FEP DB management and relation analysis is needed to facilitate the efficient composition of the scenarios. For this purpose, the CYPRUS program was developed, but abandoned due to suspended FEPs and scenario research. Thus, it became necessary to build a new easy-to-maintain platform that inherits the legacy of CYPRUS and reflects the latest research.

The data structure and user interface configuration were derived to develop a new platform. The new platform provides extensive data such as the assessment context, the FEP DB, the interaction between FEP contents, the relevance to other project FEPs, the influence on performance, the scenarios for the TSPA, the AMF, and the PA Data. The platform displays the long-term evolution FEPs developed by KAERI, the international and major project FEPs in table format. The correlation between FEP items is composed of a detailed interaction matrix and visualized as the chord diagram or arc diagram. The relevance and linkages between the project FEP items are mapped and presented in the form of network diagrams and network tables.

The platform designed in this study will be used to manage the FEP DB, analyze and visualize the relationship between the FEP and scenarios, and finally construct the performance assessment scenarios. It is expected that the platform itself will be used as a part of the knowledge management system and facilitate efficient collaboration and knowledge exchange among experts.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (NRF-2021M2E1A1085185).

Keywords: Repository for Spent Nuclear Fuel, Long-term Evolution, Performance and Safety Assessment, FEP, Scenario, Data Management System

Evaluation of Irreversible Cs Adsorption Capacity of Granite Origin Soil

Ayeong Lee^{1,2}, Chul-Min Chon¹, Jae Gon Kim^{1,*}, Jungho Ryu¹ and Yeongkyoo Kim²

¹Korea Institute of Geoscience and Mineral Resources (KIGAM)

²Kyungpook National University (KNU)

*jgkim@kigam.re.kr

Various radionuclides are released and contaminate soils by the nuclear accidents, nuclear tests and disposal of radioactive waste. Among radionuclides, ¹³⁷Cs is a harmful radioactive element that emits high-energy β particles and γ rays with a half-life of 30.2 years. ¹³⁷Cs is difficult to extract because it is fixed to soil particles. For the volume reduction technology development of contaminated soil, this study tried to evaluate the irreversible Cs adsorption capacity of granite-originated soil.

The soil sample used in the study was collected from C horizon of the soil developed in Mesozoic mica granite. The soil texture, mineralogy, organic content, pH, EC, cation exchange capacity (CEC), water-soluble cation and anion content of the soil samples were determined. A kinetic adsorption experiment and an isotherm adsorption experiment were performed to understand the overall Cs adsorption characteristics using ¹³³Cs. The desorption of Cs by 0.1 mM KCl was also tested for the sample spiked with ¹³³Cs and ¹³⁷Cs.

The soil sample showed a pH of 6.73, EC of 24.50 $\mu\text{S cm}^{-1}$, and CEC of 1.34 $\text{cmol}_c \text{ kg}^{-1}$, organic matter of 0.53% and sandy loam in texture. Quartz, feldspar and mica were identified as the major mineral components of bulk sample. The clay fraction consists of mica, hydroxyl-interlayer vermiculite (HIV), vermiculite and kaolinite. In the kinetic adsorption experiment, the Cs adsorption showed fast adsorption rates at the initial stage (6 hours) regardless of the ¹³³Cs concentration, and the adsorption equilibrium state was reached after 48 hours. It was the most suitable for the pseudo second-order model. The ¹³³Cs adsorption increased nonlinearly from low to high concentration, which was well match with the dual site Langmuir model. As a result of the desorption experiment, desorption was not performed up to 1.1 mg kg^{-1} in the presence of competitive ions K^+ , which is about 0.035% of CEC calculated by the isotherm model.

The adsorption of Cs was controlled by frayed edge sites (FES) at a low concentrations and by basal sites or interlayer sites at a high concentration. Irreversible Cs fixation of by FES may be contributed by mainly weathered mica, and when these minerals are separated from the granite origin soil, the possibility of reducing the contamination concentration and volume of radioactive soil waste can be expected.

Keywords: Cs, Granite Origin Soil, Weathered Mica, Cation Exchange Capacity (CEC), Frayed Edge Site (FES)

Fractal Constant Evaluation From Acoustic Signals for Repository Monitoring

Chang-Ho Hong, Ji-Won Kim, and Jin-Seop Kim*

Korea Atomic Energy Research Institute (KAERI)

*kjs@kaeri.re.kr

The integrity of the disposal repository structure must be guaranteed for few hundreds to few hundred thousand years until toxicity of radioactive waste is surely degraded. Acoustic emission (AE) method is widely utilized to evaluate the integrity of the structure because it can detect crack wave signals of the structures. It is well known that the cracking AE energy is proportional to the volume of the structure (Fractal theory). However, it is hard to destroy whole structures for obtaining AE energy. Therefore, the scaled specimens are prepared to obtain the relationship between volume of the structure and AE energy. The specimens are prepared with same of Wolsong Low and Intermediate Level Radioactive Waste Disposal Center (WLDC) silo concrete recipe. Their diameters are from 50 mm to 150 mm in each 10 mm and their heights are twice of the diameter. One set of 50 mm to 150 mm specimens (11 specimens in one set) are made in single mixers to maintain uniformity. Surface of the specimens are flatten with cement milk to prevent from applying load with eccentricity. The uniaxial compression test is performed by controlling displacement as 0.1 mm/min. The fractal constant is obtained using least square function from volume-cumulative AE energy relationship.

Acknowledgements

This work was supported by National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (2020M2C9A1062949 and 2022M2E3A3015608).

Keywords: Disposal Repository, Fractal Theory, Acoustic Emission Signal, Uniaxial Compression Test

How to Quantify Well the Pumping Rate From the Aquifer Around the Repository: An Approach

Youn-Myoung Lee*, Dong-Keun Cho, and Sung-Hoon Ji

Korea Atomic Energy Research Institute (KAERI)

*ymlee@kaeri.re.kr

Especially for near-surface repository for disposal of the low- and intermediate-level radioactive waste, safety assessment in case of inadvertent human intrusion should be handled seriously. This is because this type of incident will possibly give rise to high acute, not chronic exposure dose even though its occurrence of likelihood could higher than rather deeper geological repository for disposal of high-level radioactive waste over long time span after closure of the repository. Recently well drilling scenario for the pumping groundwater from the aquifer near the repository, among other possible inadvertent human intrusion incidents, has been popularly evaluated for the worst case due to its relatively high possibility of occurrence in parallel with normal scenarios for the nuclide transport for post-closure safety assessment of the repository. Movement of nuclide plume both in the confined and unconfined aquifer under and over a radioactive waste repository is of importance especially around an extracting well. Through this study a simple comment regarding quantification between a pumping rate from the well drilled into the aquifer as well as quantification of the plume size flowing around the well is presented. Drawdown of the well which is the change of water level of the upper water surface of the aquifer due to well pumping makes a cone of depression. And capture zone in the aquifer which is formed around the well, by which the groundwater is removed out, is the groundwater volume or area in the aquifer that is considered to contribute the extraction of the well by pumping. Usually this capture zone does not encompass the entire aquifer thickness for the partially penetrating well, which means that not all the portion of flowing groundwater through the aquifer is drawn by the well. And this capture zone does not need to coincide with the volume of the cone. Furthermore, all the nuclide plume volume is not necessarily and completely mixed with the groundwater flowing the entire aquifer. Therefore, a strategical approach might be required to grasp the aquifer portion and the plume size influenced by pumping to evaluate rather accurate radiological consequences due to the well scenario avoiding overestimation and meaningless conservatism as well, which is especially very common in the mass balance modeling e.g., by GoldSim under assumption that all the groundwater volume from the aquifer near the well extracted by the well. Although the capture zone around the well should be determined both by use of global/local groundwater flow model in the aquifer but a simple analytical model could be sought. Capture zone analysis has been widely seen in the area of the design of groundwater remediation system. If for safety assessment of the subsurface repository the plume behavior in the aquifer under the repository should be well characterized and correctly modeled, then the current study is expected to be more or less helpful to develop a specific mass balance model for nuclide transport and groundwater flow for assessment of an abnormal well drilling scenario near the repository.

Keywords: Safety Assessment, Repository, GoldSim, Well Scenario, Nuclide Transport, Groudwater Flow

Case Study on In-situ Inspection and Management System for Excavation-Damaged Zone Around the Disposal Caverns in a Spent Nuclear Fuel Repository

Hyung Mok Kim^{†,*}, Myung Jin Nam[†], Hanna Jang[†], Inseok Joung[†], Eui Seob Park², and Byung Gon Chae²

[†]Sejong University

²Korea Institute of Geoscience and Mineral Resources (KIGAM)

*hmkim@sejong.ac.kr

Withdrawal

4분과

중저준위폐기물관리(Oral)

The Effect of Isosaccharinic Acid (ISA) on the Solubility of Rhenium(IV) Oxide

Kyungwon Kim and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongum@postech.ac.kr

Technetium (Tc) is a long-lived radioactive isotope, which exists as TcO_4^- with high solubility under oxidative condition. The solubility of Tc is fundamental to assess the safety of radioactive waste repository in the case of a leakage of radioactive wastes. Cellulosic materials (paper, wood, cotton, etc.) contaminated by radionuclides are disposed of in low-level and intermediate-level radioactive waste repositories. Cellulose can be decomposed under anaerobic and alkaline conditions when cement pore water is saturated, and then isosaccharinic acid (ISA) is generated as a degradation product of cellulose. ISA forms complexations with radionuclides in solution and affects the solubilities of radionuclides. Therefore, the effect of ISA should be accurately evaluated to predict and assess the mobility of radionuclides in repository environments. In this study, batch tests were conducted to confirm the effect of ISA on the solubility of Rhenium(IV) Oxide. Herein, rhenium was used as a non-radioactive analog of Tc due to their similar chemical properties. Deionized water (DIW) and 0.1 M NaOH solution in pH 12.5 were used as background solutions, and ISA concentration was varied to 1~20 mM using $\text{Ca}(\text{ISA})_2$ and NaISA, respectively. The batch tests were conducted under both aerobic and anaerobic conditions. The whole batch tests under anaerobic conditions were performed in the glove box using oxygen purged DIW with a high purity nitrogen gas (99.9%) and low oxygen concentration (< 0.5 ppm). As a result, the rhenium concentration decreases as more ISA is dissolved in the solution, which shows the contrary effect of ISA on the solubility of other metals and radionuclides (e.g., Co, Th, Fe, Ni, etc.). It is assumed that the reducing capacity of ISA decreases the rhenium dissolution in the solution. Additional characterization of the oxidation state of rhenium oxide and the mechanism will be tested and presented.

Keywords: Solubility, Rhenium, Rhenium Oxide, Isosaccharinic Acid

A Study on the Treatment of Boron-Containing Discharge Water From Nuclear Power Plant Using 1 m³/hr Scale of SD-ELIX System

Seongjoo Kang, Seonggeon Kim, Sangwook Lee, Jiyang So, Ducheol Kim, Minho Kwak, and Seungil Kim*

Elim-Global

*seungil.kim@elim-global.com

Regulations on the concentration of boron discharged from industrial facilities, including nuclear power plants, are increasingly being strengthened worldwide. Since boron exists as boric acid at pH 7 or lower, it is very difficult to remove it in the existing LRS (Liquid Radwaste System) using RO and ion exchange resin. As an alternative technology for removing boron emitted from nuclear power plants, the electrochemical boron removal technology, which has been experimentally applied at the Ringhal Power Plant in Sweden, was introduced in the last presentation.

In this study, the internal structure of the electrochemical module was improved to reduce the boron concentration to 5 mg/L or less in the 50 mg/L level of boron-containing waste liquid. In addition, the applicability of the electrochemical boron removal technology was evaluated by increasing the capacity of the unit module to 1 m³/hr in consideration of the actual capacity of the monitor tank of the nuclear power plant. By applying various experimental conditions such as flow rate and pressure, the optimum boron removal conditions using electrochemical technology were confirmed, and various operating conditions necessary for actual operation were established by configuring a concentrated water recirculation system to minimize secondary waste generation. The optimal arrangement method of the 1 m³/hr unit module developed in this study was reviewed by performing mathematical modeling based on the actual capacity of monitor tank and discharge characteristics of nuclear power plant.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2019 1510301070).

Keywords: Boric acid, Boron, Electrochemical, SD-ELIX, 1 m³/hr, Scale-up

Review of the Radioactive Waste Acceptance Criteria of the Spent Filter Cartridge Both Home and Abroad for Its Improvement

Suk Nam Lim^{1,*}, Hyun Woo Song¹, Moonoh Kim¹, Jisoo Yoon², and Ji-Hoon Lee²

¹KONES Corp.

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*limsm53@kones21.com

There are generally two kinds of spent filter; one is spent filter media for mainly gaseous purification such as HEPA filter, the other is spent filter cartridge for liquid purification such as CVCS BRS cartridge type filter. The spent filter cartridge from liquid purification system has been storing in special shielding space in auxiliary building in NPPs since the beginning of 2006 according to the long term storage strategy for decaying short lived radionuclide and gaining the time for selecting practical treatment technology before final packaging.

The spent filter cartridges generated Kori-1 reactor vary in their sizes as in length from 913 mm to 290 mm and range in radiation level from several hundred mSv per hour to below mSv per hour. It is high time that the spent filter cartridge is treated and packaged because LILW repository in Wolsung area is operating and Kori-1 reactor is scheduled to decommission. The spent filter cartridge is one of the wet solid wastes required of solidification. It is difficult for the spent filter cartridge to solidify because of their shape, structure, physical and chemical characteristics in addition to having high radiation level.

NSSC notice defines that solidification of wet solid wastes include that solid material such as spent filter is encapsulated with cement, etc. as a form of macro-encapsulation. The radioactive waste acceptance criteria describes that non-homogeneous waste having above 74,000 Bq/g such as spent filter, dry active waste should be encapsulated with qualified material. Homogeneous waste such as spent resin, sludge, concentrated waste (liquid waste evaporator bottoms), etc. should be solidified complied with requirements except that spent filter which is allowed to encapsulate.

It is needed to guide to the practice of these two requirements for spent filter. The sampling and test method is different between homogeneous solidification waste form and spent filter cartridge encapsulation waste form. For example, how core sample can be taken and how void space can be measured among spent filter cartridge in encapsulation waste form.

The technical evaluation report for spent filter cartridge polymer encapsulation by US NRC has been reviewed and the technical position of US NRC was identified. As a result of review, improvement fields of waste acceptance criteria for spent filters are pointed out, and the technical position of US NRC for spent filter cartridge solidification is summarized. The recommendation on improvement directions for spent filter cartridge encapsulation is suggested.

Keywords: Spent Filter Cartridge, Solidification, Waste Acceptance Criteria, LILW, Disposal

Development of High Performance Cement Materials and Solidification Technology for Immobilization of Liquid Radioactive Wastes

Gyeong Il Moon*,¹, Seong Ho Yoo¹, Ho Yeon Yang¹, and Dong Hoon Oh²

¹Korea Environment Testing and Analysis Laboratory

²Jung Seok Chemical Corporation

*gimoon@keta.re.kr

The number of nuclear power plants that are permanently shut down or decommissioned is increasing worldwide, and accordingly, research is being conducted on an appropriate method for disposing of radioactive waste generated during the decommissioning of nuclear power plants. In the case of waste liquid generated during the decommissioning of nuclear power plants, it is important not only to efficiently reduce waste but also to secure the suitability of disposal.

One of the solidification treatment methods for radioactive waste is cement solidification, but since cement solidification has poor solidification properties and generates a large amount of waste, improvement activities have been pursued. This study aims to develop high-performance cement-based materials and solidification treatment technology for solidification of liquid radioactive waste generated during nuclear decommissioning in order to improve the problems of cement solidification treatment method.

For the development of polymer cement, epoxy resin and polyamine/amide mixed type and general Portland cement were mixed in various ratios. The most appropriate mixing ratio was 4.5:2, which showed the highest compressive strength. A simulated waste liquid was prepared by referring to the preliminary decommissioning plan of Shin-Kori Units 5 and 6, and it was dried and made into granules. Polymer cement was injected into a drum filled with granules by vacuum pressure to prepare a waste form matrix. In the solidification process, granules made by drying the waste liquid were used, and the solidification agent was filled in between the granules, so the total volume of solid radwaste was reduced compared to the conventional cement solidification treatment method. As a result, the amount of waste decreased to about 1/3, and the volume reduction rate increased by about 2.2 times. The compressive strength of 3,243 psi was confirmed in the disposability performance test for the manufactured solid samples. The compressive strength after the thermal cycling test, irradiation test, microorganism test, and immersion test was 2,257 psi, 2,306 psi, 4,530 psi, and 2,263 psi, respectively, exceeding the acceptance criteria of 500 psi. The leaching index was 7~13, and no free standing water was generated.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 3210100060).

Keywords: Polymer Cement, Solidification Technology, Suitability of Disposal

A Study on Disposal Suitability of Hanul Glass Fiber Packaging Bag

Na Rae Lee^{1,*}, Moonoh Kim¹, Sang June Park², Sung-Jun Kim², and Suil Bang²

¹KONES Corp.

²Korea Hydro & Nuclear Power (KHNP)

*nrlee@kones21.com

Currently, Hanul NPP packages glass fiber classified as particulate waste in plastic packaging bags and stores them in 200 L drums. KORAD's Waste Acceptance Criteria (WAC) presents that very low-level soil can be immobilized by loading it in a soft bag and then packaging it in a 200 L or 320 L steel drum. As currently accepted method of packaging with soft bag applies to only very low-level soils among the wastes with a risk of dispersion, it is necessary to develop a non-dispersible treatment suitable for the characteristics of other particulate waste in the future. Therefore, in order for Hanul packaging pack to be approved as an alternative method for immobilization of dispersible substances, it is necessary to verify the suitability of the packaging bag. In this paper, whether the glass fiber packaging bag used in Hanul NPP satisfies the characteristic of the soft bag presented in the WAC and the possibility of being considered as a non-dispersible measure for particulate are examined. The soft bag must meet the following requirements: material and structure, shape, drop test, and immersion test. The results of the review are as follows. First, since the glass fiber is already packaged in the drum, only the role of the inner layer, made of polyethylene, having a watertight function may be required. Second, when packaging a drum, the packaging bag is compressed into a shaped frame having an inner size of a 200 L drum, so it is packaged with little empty space in the drum. Third, as a result of a drop test of a packaging pack containing 20 kg of contents from a height of 1.2 m, it was confirmed that there was no leakage of contents. Fourth, the packaging bag was immersed in a 1-m depth water tank for 30-minutes, and the performance corresponding to the IPX7 was satisfied. As a result of reviewing the soft bag characteristic of Hanul glass fiber packaging bag, it is considered that the bag can be used as one of the non-dispersible measures because it meets almost the characteristics required by the WAC. In addition, the acceptance criteria of overseas disposal sites present various secure packaging methods in place of immobilization as a non-dispersible measure for waste containing particulate matter. It is necessary to reflect these overseas cases in the establishment of non-dispersible measures for domestic waste acceptance in the future.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. 20211510100040).

Keywords: Glass Fiber, Packaging Bag, Soft Bag, Non-dispersible Measures, Waste Acceptance Criteria

A Study on the Carbon Dioxide Adsorption Characteristics Using Radioactive Waste Soil/Waste Concrete

Sangwook Lee, Seongjoo Kang, Duchul Kim, Jiyang So, Seonggeon Kim, Minho Kwak, and Seungil Kim*

Elim-Global

*seungil.kim@elim-global.com

In the case of decommissioning of a nuclear power plant, it is expected that a significant amount of VLLW and LLW that need to be disposed of are also expected. Conventional reduction technology is a method of extracting or removing radionuclides from waste, but this project is being carried out for the purpose of obtaining a reduction effect through the development of a material that treats another radioactive waste using radioactive waste.

In this paper, the technology of impregnating LiOH capable of adsorbing radiocarbon to the gas filter material manufactured from concrete and soil waste as raw materials and the radiocarbon removal performance were reviewed.

In this study, a raw material of ceramic filter was prepared by mixing concrete and soil waste with a powder of 40 μm or less, and after sintering at 1,250°C, 5wt% to 40wt% of LiOH is impregnated with a filter capable of adsorbing carbon dioxide. was prepared. The prepared filter used ICP-OES and XRD to confirm the LiOH deposition result, and the concentration of carbon dioxide discharged through the carbon dioxide adsorption device was confirmed. It was possible to obtain the result that the amount of adsorption was changed depending on the flow rate of carbon dioxide supplied and the amount of material. Through this, it was possible to confirm the possibility of power generation in the adsorption performance of gas.

In this study, after crushing waste concrete and waste soil, powders of 40 μm or less were mixed with other additives to prepare raw materials for ceramic filters, and sintered at 1,250°C to manufacture filters. 5wt% to 40wt% of LiOH was impregnated on the prepared filter to give functionality to enable carbon dioxide adsorption. The results of LiOH deposition were confirmed using ICP-OES and XRD, and the change in the concentration of carbon dioxide emitted through a separately prepared adsorption device was confirmed. It was possible to obtain the result that the amount of adsorption was changed according to the flow rate of carbon dioxide supplied and the amount of material, and the possibility of developing a material for radioactive waste treatment using radioactive waste was confirmed when the porosity and specific surface area of the filter material were increased.

Acknowledgements

Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 20201520300070).

Keywords: Waste Soil, Waste Concrete, Carbon Dioxide, LiOH, Radioactive Gases

Preliminary Safety Assessment of Worker of Glass Fiber Waste Under Accident Conditions at Disposal Facility

Woo-Yong Kim^{*1}, Sang June Park², Sungjun Kim², Suil Bang², and Moonoh Kim¹

¹KONES Corp.

²Korea Hydro & Nuclear Power (KHNP)

*kwy@kones21.com

Glass fiber (GF) insulation is a non-combustible material, light, easy to transport/store, and has excellent thermal insulation performance, so it has been widely used in the piping of nuclear power plants. However, if the GF insulation is exposed to a high-temperature environment for a long period of time, there is a possibility that it may be crushed even with a small impact due to deterioration phenomenon and take the form of small particles. In fact, GF dust was generated in some of the insulation waste generated during the maintenance process.

In the previous study, the disposal safety assessment of GF waste was performed under the abnormal condition of the disposal facility to calculate the radiation exposure dose of the public residing/residents nearby facilities, and then the disposal safety of GF waste was verified by confirming that the exposure dose was less than the limit. However, the revised guidelines for safety assessment require the addition of exposure dose assessment of workers. Therefore, in this study, accident scenarios at disposal facilities were derived and the exposure dose to the workers during the accident was evaluated.

The evaluation was carried out in the following order: (1) selection of accident scenario, (2) calculation of exposure dose, (3) comparison of evaluation results with dose limits, and confirmation of satisfaction. The representative accident scenarios with the highest risk among the facility accident were selected as; (a) the fire in the treatment facility, (b) the fire in the storage facility, and (c) fire after a collision of transport vehicles. The internal and external exposure doses of the worker by radioactive plume were calculated at 10m away from the accident point. In evaluation, the dose conversion factors ICRP-72 and FGR12 were used. As a result of the calculation, the exposure dose to workers was derived as about 0.08 mSv, 0.20 mSv, and 0.10 mSv, due to fire accidents (vehicle collision, storage facilities, treatment facilities). These were 0.2%, 0.4%, and 0.2% of the limit, and the radiation risk to workers was evaluated to be very low.

The results of this study will be used as basic data to prove the safety of the disposal of GF waste. The sensitivity analysis will be performed by changing the radiation source and emission rate in the future.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. 20211510100040).

Keywords: Radioactive Waste, Safety Assessment of Disposal, Glass Fiber Waste, LILW Disposal Facility, Dose Limit, Waste Acceptance Criteria

Fabrication of Cs Pollucite Ceramic Waste Form and Long-term Leaching Test for Immobilization of Radioactive Cesium

Ga Yeong Kim^{1,2,*}, Ki Rak Lee², Jung Hoon Choi², Jae Hwan Yang¹, and Hwan Seo Park²

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*kky@kaeri.re.kr

Radioactive cesium is a heat generated and semi-volatile nuclide in spent nuclear fuel (SNF). It is released as a gaseous phase by head-end treatment which is a pretreatment of pyroprocessing. One of the capturing methods of gaseous radioactive cesium is using zeolite. After ion-exchanged zeolite, it is transformed to ceramic waste form which is a durable ceramic structure by heat treatment. Various ceramic wasteforms for Cs immobilization have been researched such as cesium aluminosilicate ($\text{CsAlSi}_2\text{O}_6$), cesium zirconium phosphate ($\text{CsZr}_2(\text{PO}_4)_3$), cesium titanate ($\text{Cs}_x\text{Al}_x\text{Ti}_{8-x}\text{O}_{16}$, $\text{Cs}_2\text{TiNb}_6\text{O}_{18}$) and $\text{CsZr}_{0.5}\text{W}_{1.5}\text{O}_6$. The cesium pollucite is composed of an aluminosilicate framework and cesium ions incorporated in matrix materials lattices. Many researchers are reported that the pollucite has high chemical durability. In this study, the cesium pollucite was fabricated using mixtures of aluminosilicate denoted as absorbent product (AP) and Cs_2CO_3 by calcination and pelletized by cold pressing. The characterization of fabricated pollucite powder and pellets was analyzed by XRD, TGA, SEM, SEM-EDS and XRF. The chemical durability of pollucite powder was evaluated by PCT-A and ICP-MS and OES. Thus, the optimal pressure condition without breaking the pellets which is a low $\text{Cs}_2\text{O}/\text{AP}$ ratio and pelletizing pressure was selected.

The long-term leaching test was performed using MCC-1 method for 28 days with the fabricated pollucite pellets. The leachate of leaching test was allard groundwater and Deionized water and replaced 5 contact periods which are 3 hours, 3 days, 7 days, 14 days and 28 days and analyzed by ICP-MS. The leaching rate was shown in two stages. The first stage was rapid and a relatively large amount of nuclides were leached. The leaching rate was decreased in the second stage. The fractional release rate of this study was shown the same trend. These results were similar to previous studies.

Keywords: Radioactive Cesium, Cesium Pollucite, Ceramic Waste Form, Leaching Test

4분과

중저준위폐기물관리(Poster)

Study on Colloidal Impact of Glass Fiber Particle Considering Sorption Ability

Hyun Woo Song^{1,*}, Moonoh Kim¹, Sang June Park², Sungjun Kim² and Su-il Bang²

¹KONES Corp.

²Korea Hydro & Nuclear Power (KHNP)

*shw@kones21.com

Colloid migration is an important topic in post-closure safety assessment of radioactive waste repository as radionuclide can be adsorbed onto colloidal particles and migrated along with the colloids. This would reduce retardation of radionuclide migration, thus increasing the released concentration into biosphere. Recently, glass fiber waste has been found to contain small sized crushed glass fiber particles (GFPs), and concerns regarding the colloidal impact of GFP is being discussed. In this study, relevance of assessing GFPs facilitated radionuclide transport in the disposal environment of 1st phase disposal facility.

Colloidal impact assessment can be divided into two sections, colloid mobility, and colloid sorption assessments. Considering GFP being denser than water, fluid velocity of 1st phase disposal facility is too slow to initiate movement of such dense particles. GFPs would remain settled, and no colloidal impact is expected. In this study, sorption assessment mainly focused to analyze the possible impact if migration of GFP does occur.

The GFP is mainly composed of SiO₂ and few other metal oxides. Due to high composition of SiO₂ in the GFPs, negative surface charge is induced onto the surface of the GFPs in alkaline environment. This negatively charged surface can attract free positive ions (ex. Ni, Co, Fe, etc.) in the repository, and these ions would be adsorbed onto the surface of the GFPs via coulomb force. Thus, if GFPs migrate, colloid facilitated radionuclide transport can be expected.

However, before being released into the biosphere, particles must pass through the engineered and natural barriers, where ion-colloid-rock interactions could result in transfer of radionuclide from one media to another. At Naka Research Center, Japan, ion-colloid-rock interactions are experimented with bentonite colloid, and the result showed that despite colloid's sorption ability was 10 times higher than the barrier material, the overall released radionuclide concentration has negligible change. To reflect such phenomenon, coulomb attractive force of GFPs and concrete is calculated and compared, which the result showed that glass fiber was 10 times weaker than concrete. Considering the Japan's experimental result, glass fiber facilitated transport would not enhance the radionuclide release into the biosphere.

Nonetheless, assuming GFPs being mobile in 1st phase disposal facility, GFPs' sorption ability is found to be negligible compared to the concrete of the repository, thus radionuclide transport is not expected to be enhanced. In future, this study could be used as basis for further colloidal impact analysis for the safety assessment of the repository.

Acknowledgements

This work is supported by KOREA HYDRO & NUCLEAR POWER Co., Ltd (No. E21S063000).

Keywords: Glass Fiber, Colloidal Impact, Sorption Ability, Disposal Environment, Post-Closure Safety Assessment

Evaluation of Leaching Behavior and Thermal Stability of Silver Tellurite Glass

Seong Sik Shin^{1,2,*}, Hyun Woo Kang², Jung-Hoon Choi², Ki Rak Lee², and Hwan-Seo Park²

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*seongsik2076@kaeri.re.kr

Uranium-235, used in nuclear power generation, produces a lot of radioactive waste. Among radioactive waste nuclides, I-129 is problematic due to its long half-life (1.57×10^7 y) with high mobility in the environment. It should be captured and immobilized into a geological disposal environment through a stable waste form. In this study, various additives including Al, Bi, Pb, V, Mo and W were added to silver tellurite glass to prepare a matrix for immobilizing iodine, and its thermal and leaching properties were evaluated. To prepare glass, the glass precursor mixture was placed in alumina crucibles and heated at 800°C for 1 h. Except for aluminum, there was no significant loss of constituent elements. The loading of iodine in the matrix was approximately 11-15% by weigh, excluding oxygen. The normalized releases of all the elements obtained by PCT-A were below the order of 10^{-1} g/m², which satisfies US regulation (2 g/m²). Differential scanning calorimetry was performed to evaluate the thermal properties of the glass samples. The glass transition temperature (T_g) increased by adding such as V₂O₅, MoO₃, or WO₃. The similar relative electrostatic field values of V₂O₅, MoO₃, and WO₃ could provide sufficient electro static field to the TeO₂ interacting with the non-bridging oxygen forming Te-O-M (M = V, Mo, W) links. The addition of MoO₃ or WO₃ in the silver tellurite glass system increased glass transition temperature (T_g) and crystallization temperature (T_c) while maintaining the glass stability.

Keywords: Radioactive Waste, Iodine-129, Silver Tellurite Glass, Thermal Stability, Leaching Property

A Design of Optimal Repackaged Algorithm for Reduction of Low-and Intermediate-level Radioactive Waste

Hee-Seoung Park*, Il-Sik Kang, Dong-ju Lee, Choong-Wie Lee, Won-Hyuk Jang, Jong-Jin Kim,

and Jin-Woo Lee

Korea Atomic Energy Research Institute (KAERI)

*parkhs@kaeri.re.kr

Korea Radioactive Waste Agency (KORAD), regulatory body and civic groups are calling for an infrastructure system that can more systematically and safely manage data on the results of radioactive waste sampling and nuclide analysis in accordance with radioactive waste disposal standards. To solve this problem, a study has been conducted on the analysis of the nuclide pattern of radioactive waste on the nuclide data contained in low-and intermediate-level radioactive waste. This paper will explain the optimal repackaged algorithm for reducing radioactive waste based on previous research results.

The optimal repackaged algorithm for radioactive waste reduction is comprised based on nuclide pattern association indicators, classification by nuclide level of small-packaged waste, and nuclide concentration. Optimization simulation is carried out in the order of deriving nuclide concentration by small-packaged, normalizing drum minimization as a function of purpose, normalizing constraints, and optimization. Two scenarios were applied to the simulation. In Scenario 1 (generating facilities and repackaged by medium classification without optimization), it was assumed that there are 886 low-level drums and 52 very low-level drums. In Scenario 2 (generating facilities and repackaged by medium classification with optimization), 708 and 230 drums were assigned to the low-level and very low-level drums, respectively. As a result of the simulation, when repackaged in consideration of the nuclide concentration and constraints according to the generating facility cluster & middle classification by small package (Scenario 2) the low-level drum had the effect of reducing 178 drums from the baseline value of 886 drums to 708 drums. It was found that the reduced packages were moved to the very low-level drum.

The system that manages the full life-cycle of radioactive waste can be operated effectively only when the function of predicting or tracking the occurrence of radioactive waste drums from the source of radioactive waste to the disposal site is secured. If the main factors affecting the concentration and pattern of nuclides are systematically managed through these systems, the system will be used as a useful tool for policy decisions that can prevent human error and drastically reduce the generation of disposable drums.

Acknowledgements

This study was supported by the Nuclear Research R&D Program (2019M2C9A1059067) through the National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT (MSIT), Republic of Korea.

Keywords: Reduction of Radioactive Wastes, Optimal Repackaged Algorithm, Nuclide Pattern Association Indicators, Small-packaged Waste

Solubility of Radionuclides Under Near and Far Field Conditions: Effect of Organic Complexing Agents

Bolam Kim, Ahsan Abdul Ghani, Youngsu Lim, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Organic complexing agents which are contained in the radioactive waste can form the complex with radionuclides and enhance the solubility of radionuclides. The mobility of radionuclides to the far-field from the repository will be increased by radionuclide-ligand complex formation. Therefore, the assessment of the radionuclides' solubility should be performed in the presence of organic complexing agents. In this study, five radionuclides (cobalt, strontium, iodine, cesium, and uranium) and three organic complexing agents (ethylenediaminetetraacetic acid (EDTA), nitrilotriacetic acid (NTA), and isosaccharinic acid (ISA)) were selected as model radionuclides and organic complexing agents, respectively. For simulating the in-situ condition, the groundwater near the repository was collected and applied in solubility experiments and the solubility was measured in various environmental conditions such as different pHs (7, 9, 11, and 13), temperatures (10°C, 20°C, and 40°C), and a range of organic complexing agent concentrations (10^{-5} , 10^{-4} , 10^{-3} , and 10^{-2} M). In cases of cesium and iodine, they were very soluble in all conditions, and the effect on their solubilities was not observed. However, at high pHs, cobalt and strontium showed lower solubilities than at neutral pH and the solubility enhancement by the organic complexing agents was significant. Moreover, the effects of each organic ligand showed obvious differences and were in the order of EDTA > NTA > ISA. The solubility of uranium was increased with increasing the organic ligand concentration at lower pHs, but the organic complexing agents did not cause a remarkable difference at high pHs. According to these results, the presence of complexing agents could enhance the radionuclides' solubility and increase the potential to release the radionuclides to the far-field from the repository. Solubility experiments of other major radionuclides in the repository are in progress.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Solubility, Radionuclide, Organic Complexing Agent, Radionuclide-Ligand Complex

A Study on Low-Radiation Waste Resin Treatment Using Plasma Torch Melting Technology

Jeongsu Jeong, Seoyong Choi, Sunghoon Hong*, and Mingyu Cho

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sunghoon.hong@khnp.co.kr

The liquid radioactive waste system of nuclear power plants treats radioactive contaminated wastes generated during the Anticipated Operational Occurrence (AOO) and normal operation using filters, ion exchange resins, centrifuges, etc. When the contaminated waste liquid is transferred to an ion exchanger filled with cation exchange resin and anion exchange resin, nuclides such as Co and Cs are removed and purified. The lifespan and replacement time of the ion exchange resin are determined by performing a performance test on the sample collected from the rear end of the ion exchanger, and waste ion exchange resin is periodically generated in nuclear power plants. In the general industry, most waste resins at the end of their lifespan are incinerated in accordance with related laws, but waste resins generated from nuclear power plants are disposed of by clearance or stored in a HIC (High Integrity Container). Plasma torch melting technology can reduce the volume of waste by using high-temperature heat (about 1,600 degrees) generated from the torch due to an electric arc phenomenon such as lightning, and secure stability suitable for disposal. Plasma torch melting technology will be used to check thermal decomposition, melting, exhaust gas characteristics, and volume reduction at high temperatures, and to ensure disposal safety. Through this research, it is expected that the stable treatment and disposal of waste resins generated from nuclear power plants will be possible.

Keywords: Plasma Torch Melting Technology, Resin, Liquid Radioactive Waste, Demonstration Test

A Parametric Study on Shadow Zone Effect in Near-Field Radionuclide Transport Modeling for an Engineered Vault Type LILW Disposal Facility

Wontak Lee, Dongki Kim, and Joowan Park*

RADCORE

*p5j9w@radcore.co.kr

The “shadow zone” is defined as a region below a flow obstacle, such as a vault, in unsaturated soils. Due to the capillary discontinuity of the cavity, water saturation on the top and side of the cavity is higher than the ambient saturation. On the bottom of the cavity, however, there is a region where water saturation is lower than ambient saturation. Undoubtedly, a shadow zone may also exist below a LILW disposal vault built in subsurface soils above the water table before the vault is fully degraded. During the degradation, flow in the shadow zone is controlled by the rate of water infiltrating the degrading vault.

In this study, as one of the efforts to be made for enhancing safety margin by a realistic safety assessment of the engineered vault type LILW disposal facility, the shadow zone effect is investigated by a numerical parametric study using AMBER code. The conceptual model and data were excerpted from IAEA, ISAM Vault Test Case for the liquid release design scenario. It is assumed that the near-field barriers degrade with time. In order to compare a visible shadow zone effect, the vault degradation period is assumed to be both 500 and 1,000 years, and the shadow zone depth to be varied according to unsaturated zone lithology.

It can be seen that with a shorter shadow zone (2.7 m), radionuclides arrive at the water table earlier than with a full shadow zone (55 m) due to increased advection rate in the unsaturated zone. This effect tends to be more visible in the case of a longer degradation period. For radionuclides with short residence time relative to their half-lives in the unsaturated zone, such as Tc-99 and I-129, the radionuclides are shown to come out because they will arrive sooner, thereby allowing less peak release rate, when the shadow zone effect is considered. Once the vault is completely degraded and the infiltration rate of water flowing through the vault is equal to the ambient rate, the shadow zone effect disappears.

In this example calculations using IAEA ISAM Vault Test Case input parameters, it might not be shown a significant shadow zone effect. Nevertheless, when the extent of the shadow zone is determined through more sophisticated hydraulic studies in the unsaturated soils surrounding the vault, the shadow zone effect would be checked up on the realistic near-field radionuclide transport modeling in order to contribute to gaining safety margins for post-closure safety assessment of the Wolsong 2nd phase LILW disposal facility.

Keywords: Shadow Zone Effect, Unsaturated Zone, Engineered Vault Type Disposal Facility, Near-Field Transport Modeling

Development Trend of Radwaste Characterization Technology for Safe Disposal

Seok-Ju Hwang*, YoungHwan Hwang, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hsjhhhhh@khnp.co.kr

For the disposal of radioactive waste generated from nuclear power plants, characterization of radioactive waste is essential. For characterization, samples of radioactive waste are directly collected or an indirect method is used through X-ray, etc.

Through indirect analysis, which is a non-destructive method, the density, filling height, homogeneity and inter structure of the waste container can be analyzed. Currently, foreign institutions are in the process of developing a technology to perform characterization of radioactive waste through indirect analysis. In particular, research on improving internal image accuracy through image analysis techniques, improving measurement methods and enhancing portability for field application is ongoing. Through the review of such technology development trends, it will be utilized in the development of domestic radioactive waste disposal technology.

Keywords: Characterization, Radwaste, Disposal, X-ray Imaging

Degradation of Cellulosic Materials Into Isosaccharinic Acid Both Under Partially Aerobic and Completely Anaerobic Conditions

Bolam Kim¹, Jiseon Jang², Seho Choi², Myunggoo Kang², Jaechul Ha², Chun-Hyung Cho², Sanghwa Oh¹,
and Dae Sung Lee^{1,*}

¹Kyungpook National University (KNU)

²Korea Radioactive Waste Agency (KORAD)

*daesung@knu.ac.kr

Cellulose-based wastes can be degraded into short-chain organic acids at the cementitious radioactive waste repository. Isosaccharinic acid (ISA), one of the main degradation products, can form the chelate complex with metals and radionuclides, and these complexes have a potential that can accelerate to move the radionuclides to far-field from the repository. This study characterized the amount of generated ISA from typical cellulosic materials in the repository. Two different degradation experiments were conducted under alkaline conditions (saturated with $\text{Ca}(\text{OH})_2$ at pH 12.4): i) cellulosic material mixture under an opened condition (partially aerobic), and ii) cellulosic material under an anaerobic condition in a nitrogen-purged glove box. In the first case, three different types of cellulosic materials—paper, cotton, and wood—were mixed at the same ratio, and the experiments were carried out at three different temperatures (20°C, 40°C, and 60°C). It revealed that both the cellulose degradation rate and generated ISA concentration were high at high reaction temperatures, and various soluble degradation products such as formic acid and lactic acid were generated. The cellulose degradation in this work seems to still stay at a peeling-off process. In the second study, each type of cellulosic material was applied in its own batch experiments, and the amount of generated ISA was in the order of paper > wood > cotton. The above two experiments are supposed to be a long-term study until the generated ISA reaches an equilibrium state.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Cellulose Degradation, Alkaline Condition, Isosaccharinic Acid, Radionuclide-ISA Complex

Air Conditioning Condensate Contamination From Radiation Management Area of the DU Facility

Yongsang Lee*, Youngwoo Kwon, and Seonho Noh

Korea Atomic Energy Research Institute (KAERI)

*ys2@kaeri.re.kr

The Nuclear Cycle Experiment Research Center is one of the facility of the Korea Atomic Energy Research Institute (KAERI). This facility is a laboratory-scale version of pyro-processing technology. Mixture depleted Uranium (DU) and depleted Uranium (DU) feed material are used in this facility for pyro-research. During summer, air conditioners that maintain temperature and humidity are always in operation to protect analysis equipments. 15 air conditioners are installed in this facility. The condensate which is generated in 15 air conditioners is collected in one place to analyze. Sampling was performed to check the level of contamination, U, pH and gamma radiation test were performed.

This paper shows the degree of contamination of air conditioner condensate which is generated in the radiation management area.

Keywords: DU, Depleted Uranium, Airconditioning Condensate, Radioactive Waste

Analysis of Methodology for Deriving the Total Amount of SKB Cellulose-Containing Radioactive Waste

Bora Yang, Jaechul Ha*, Myunggoo Kang, and Seho Choi

Korea Radioactive Waste Agency (KORAD)

*jcha@korad.or.kr

Radioactive waste containing cellulosic materials such as cotton, paper and wood are being disposed in Low-and intermediate-level radioactive waste disposal site in Gyeongju. Cellulose has recently emerged great issue in terms of disposal site safety as it can be decomposed into an organic complex compound, ISA (isosaccharinic acid), under strong alkali conditions (pH 12.5 or higher) formed by the hydrated cement, to accelerate the mobility of the radionuclides in the disposal facility. However, in Korea, there are insufficient criteria for confirming the suitability for disposal of low-and intermediate-level radioactive wastes including cellulose, and there is no specific method for evaluating the total amount of waste to confirm the suitability of disposal. Therefore, the method of SKB (Swedish Nuclear and Fuel Management Company), which has established acceptance criteria related to the physical-chemistry safety of cellulose, is analyzed to suggest a method for deriving the amount of cellulose-containing waste disposal.

Cellulose, an organic complexing agent, is an important consideration for safety case at the Swedish low-and intermediate-level radioactive waste disposal site SFR. SKB calculated the amount of cellulose generated by separately labeling cellulose-containing wastes of 1-2BMA, Silo and 1BTF (SKB 2013). BLA, a low-level radioactive waste disposal facility, is not considered due to its low radionuclide inventory (~0.2% of SFR's total radionuclide inventory, SKB 2013).

To calculate the amount of cellulose that can be disposed of, information on the mass and volume of hydrated cement (concrete waste, cement solidification waste, disposal container, grouting, disposal shed), the concentration of ISA absorbed in the hydrated cement, and the concentration of ISA dissolved in the groundwater which were used. In addition, the total disposable amount was calculated using the cellulose degradation rate, composition ratio, and the cellulose containing waste volume.

Keywords: Organic Complex Compound, Organic Complexing Agent, ISA, Cellulose

Optimization of Reactor Vessel Internals Segmentation of Kori Unit 1

Kyu-Sang Hwang* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*hksjin@naver.com

It is necessary to prepare for cutting and storing waste materials in the reactor vessel internals (RVI) for successful decommissioning of the nuclear power plant (NPP). Since RVI contain massive components and relatively highly activated, their decommissioning process should be conducted carefully in terms of radiological and industrial safety. To achieve efficient decommissioning waste management, this study presents radiation level of RVI and cutting optimization was performed for intermediate level waste.

As a result of the radiation evaluation, a part of the core side and the upper part of RVI were evaluated as intermediate-level waste, and other components were evaluated as very low-level or low-level waste. For intermediate-level waste cutting, the minimum cutting method that can be put into a container was reviewed in consideration of the size, thickness, and cutting method of the interior product. The final segmentation parts are expected to be loaded into two storage containers.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP Project No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Decommissioning, Segmentation, Reactor Vessel

^{239}Pu and ^{99}Tc Sorption to the Cementitious and Natural Rock Barriers: Effect of Organic Complexing Agents (EDTA and ISA)

Seongju Lee, Youngsu Lim, Bolam Kim, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Radionuclides can be leached into groundwater or soil over a long period of time due to unexpected situations even after being permanently disposed of in a repository. Therefore, it is necessary to investigate the mobility of radionuclides for the safety assessment of radioactive waste disposal. In this study, the effects of organic complexing agents such as ethylenediaminetetraacetic acid (EDTA) and isosaccharinic acid (ISA) on the sorption behavior of ^{239}Pu and ^{99}Tc over cementitious (concrete and grout) and natural rock samples (granite and sedimentary rock) were investigated in batch sorption experiments. For characterization of rock samples, XRD, XRF, FT-IR, FE-SEM, BET, and Zeta-potential analyses were performed. For the evaluation of mobility, the distribution coefficient (K_d) was selected and compared. The adsorption experiment was carried out at two pHs (7 and 13), a temperature of 20°C, and a range of organic complexing agents concentrations ($10^{-7}\sim 10^{-2}$ M and $10^{-5}\sim 10^{-2}$ M for ^{239}Pu and ^{99}Tc , respectively). The radionuclides concentrations in adsorption samples were analyzed using ICP-MS. The K_d values for ^{239}Pu in all rock samples reduced significantly due to the presence of EDTA, even at low concentrations such as 10^{-5} M. In the case of ISA, the limiting no-effect concentration was much higher than that of EDTA. On the other hand, ^{99}Tc showed relatively lower K_d values than ^{239}Pu , and the sorption behavior of ^{99}Tc was almost unaffected by the organic complexing agents for all rock samples. Therefore, it is possible to assume that the increased mobility of radionuclides, especially, ^{239}Pu , in groundwater caused by the lowering of sorption at even low concentrations of organic complexing agents may result in the transport of radionuclides to the near- and far-field location of the repository.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Plutonium, Technetium, Organic Complexing Agents, Distribution Coefficient, Mobility

Cesium Adsorption on Engineered and Natural Barriers in a Fixed-bed Column System: Effects of EDTA and ISA

Juhui Park, Youngsu Lim, Bolam Kim, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Radionuclides stored in a radioactive waste repository over a long period of time might be leached through the barriers such as engineered rock (cement) and natural rock (granite). Organic complexing agents such as ethylenediaminetetraacetic acid (EDTA) and isosaccharinic acid (ISA) may also influence the mobility of radionuclides. In this study, a continuous fixed-column reactor packed with engineered and natural rocks was designed to investigate the effect of organic complexing agents on cesium mobility through cement and granite under anaerobic conditions. The influent flow rate of the mixed solution (organic complexing agent and cesium) at the column bottom was 0.1 mL/min, while that of groundwater was 0.2 mL/min, which was introduced between cement and granite layers in the middle of the column. The hydraulic properties such as diffusion coefficient and retardation factor were derived by a bromide tracer test. The effects of different operating parameters, such as initial cesium concentrations, initial EDTA or ISA concentrations, and bed size, on the cesium adsorption were investigated. The Thomas, Adams-Bohart, and Yoon-Nelson models were applied to the experimental data to predict the breakthrough curves using non-linear regression. These results suggest that organic complexing agents such as EDTA and ISA significantly influence the mobility of cesium in the barriers, indicating that the presence of complexing agents enhances the migration of cesium to the geosphere.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Cesium, Adsorption, Continuous Fixed-column Test, Tracer Test, Organic Complexing Agents

Effect of Organic Complexing Agents on Sorption and Mobility of Radionuclide ^{90}Sr by Engineered and Natural Rock Barriers at L/ILW Disposal Facility

Beomgeol Jang, Ahsan Abdul Ghani, Bolam Kim, Youngsu Lim, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Engineered barriers (concrete and grout) in Low- and Intermediate-Level Waste (L/ILW) disposal facilities tend to degrade by groundwater or rainfall water over a long period of time. During the degradation process, radionuclides stored in the disposal facility might be released into the pore water, which can pass through the natural rock barriers (granite and sedimentary rock) and may reach the near-field and far-field. In this transportation, radionuclide might be sorbed onto the engineered and natural rock barriers. In addition, the organic complexing agent such as ethylenediaminetetraacetic acid (EDTA) and α -isosaccharinic acid (ISA), is also present in pore water, which may affect the sorption and mobility of radionuclide. In this study, the sorption and mobility of ^{90}Sr under different conditions such as two pHs (7 and 13), different initial concentrations of organic complexing agents (from 10^{-5} M to 10^{-2} M), and solutions (groundwater, pore water, and rainfall water) were investigated in a batch system. The groundwater was collected at the L/ILW disposal facility located at Gyeongju in South Korea. The pore water and rainfall water were artificially made in the laboratory. The concrete, grout, granite, and sedimentary rock samples were collected from the same study sites from where the groundwater was collected. The rock samples were crushed to 53-150 micrometers and were characterized by XRD, XRF, SEM-EDS, BET, and zeta potential analyzer. ^{90}Sr concentration was determined using liquid scintillation counting. The sorption of ^{90}Sr was described by distribution coefficients (K_d) and sorption reduction factor (SRF). In the case of EDTA, the K_d values of ^{90}Sr remained constant from 10^{-5} M to 10^{-3} M and tended to decrease at 10^{-2} M, while in case of ISA the K_d values decreased steadily as the concentration of ISA was increased from 10^{-5} M to 10^{-3} M; However, a sudden reduction in the K_d values were observed above 10^{-2} M. In comparison to EDTA, ISA gave a higher SRF of ^{90}Sr . Therefore, from the above results, it can be concluded that the presence of ISA has a greater effect on the sorption and mobility of radionuclide in the solutions than EDTA, and the radionuclide may reach near- and far-field of the L/ILW disposal facility.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Strontium-90, Sorption, Mobility, Distribution Coefficients, Sorption Reduction Factor, Organic Complexing Agents

Effect of Chelating Agents and Cement Degradation States on the Transport Behavior of ^{238}U in Engineered and Natural Barriers

Youngsu Lim, Juhui Park, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Low- and intermediate-level radioactive wastes have been disposed of in the first-phase deep underground silo disposal at Gyeongju in South Korea. These radioactive wastes contain harmful radionuclides such as Uranium-238 (^{238}U), which can pose long-term and deleterious effects on humans and the natural environment. Ethylenediaminetetraacetic acid and isosaccharinic acid, which can be formed via cellulosic waste degradation under high alkaline conditions might considerably enhance the transport behavior of ^{238}U with the intrusion of rainwater and groundwater. In this study, the engineered barriers (concrete and grout) and natural barriers (sedimentary rock and granite) were used to investigate the ^{238}U transport behavior in artificial cementitious porewater of State I (pH 13.3) and State II (pH 12.5) based on groundwater or rainwater. The surface properties and geochemical compositions of barrier samples were characterized using XRD, XRF, SEM-EDX, and BET. The transport behaviors of ^{238}U in various solution conditions were observed by sorption distribution coefficient (K_d) at a range of initial chelating agents concentration (10^{-5} - 10^{-2} M). The sorption behavior of ^{238}U was retarded more in the engineered rock barriers than in the natural rock barriers. The mobility enhancement of ^{238}U was more significant in State I than in State II. In comparison with the absence of chelating agents, negligible changes in the K_d values of ^{238}U were observed at less than initial chelating agent concentrations of 10^{-4} M. However, the K_d values of ^{238}U were significantly reduced at initial chelating agent concentrations higher than 10^{-3} M. Therefore, these experimental findings show that the transport behavior of ^{238}U into the geo- and bio-sphere could be accelerated by the presence of chelating agents and the type of cement degradation states.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Uranium, Transport, Sorption Distribution Coefficient, Chelating Agents, Cement Degradation

Self-disposal of Metal Radioactive Waste

Kyeong Min Park*, Si Hyung Kim, Jong Hun Kim, Yun Jeong Hong, Seong Jae Cheon, Sunghwan Kim,
and Yong Jin Jeong

Korea Atomic Energy Research Institute (KAERI)

*pkmin@kaeri.re.kr

The fuel fabrication facility has been built and is being operated by KAERI since licensing research reactor fuel fabrication in 2004. After almost 20 years of operation, outdated equipment for fabrication or inspection has been replaced by automated, digitalized ones to assure a higher quality of nuclear fuels. However, the generation of a large amount of radioactive waste is another concern for the replacement in terms of its volume and various types of it that should be categorized before disposal. The regulatory body, NSSC (Nuclear Safety and Security Commission) released a notice related to the classification of radioactive wastes, and most accessory equipment can be classified into the clearance levels, called self-disposal waste.

In this study, the practice of self-disposal of metal radioactive waste is carried out to reduce its volume and downgrade its radioactivity. For metal radioactive waste, which is expected to occupy the most amount, analysis status and legal limitations were performed as follows: First, the disposal plan was established after an investigation of the use history for equipment. Second, those were classified by types of materials, and their surface radio-contamination was measured for checking self-disposable or not. After collecting data, the plan for the self-disposal was written and submitted to the Korea Institute of Nuclear Safety (KINS) for approval.

Acknowledgements

This research was supported by a grant from the Nuclear Energy and Safety Program of the National Research Foundation of Korea (NRF) funded by the Korean government (MSIT) (No. NRF-2021M2E5A1044516).

Keywords: Self-disposal, Radioactive Waste, Metal, Radioactivity

Tellurite Glasses for Immobilization of Selenium

Jae-Young Pyo*, Jung Hoon Choi, Hyun Woo Kang, Ki Rak Lee, and Hwan Seo Park

Korea Atomic Energy Research Institute (KAERI)

*jypyo@kaeri.re.kr

Se-79, a fission product of uranium, is present in spent nuclear fuel. Selenium is volatilized from the spent nuclear fuel during the pretreatment of pyroprocessing, and a filter composed of calcium oxide can capture gaseous selenium in the form of CaSeO_3 . Because Se-79 has a long half-life (3.27×10^5 years) and selenite ions have high mobility in groundwater, they must be immobilized in a chemically stable form for final disposal. This study used a composition of 50 TeO_2 - 10 Al_2O_3 - 10 B_2O_3 - 10 Na_2O - 10 CaO - 10 ZnO (mol%). High-purity powders of TeO_2 , Al_2O_3 , H_3BO_3 , Na_2CO_3 , CaCO_3 , and ZnO were used as glass precursors. The mixed powders were placed in alumina crucibles and melted in an electric furnace under an ambient atmosphere at 800°C for 1 h before being cast on a carbon mold. The obtained glasses were ground into fine powders and then mixed with CaSeO_3 powders. The powders were melted in alumina crucibles at 800°C for 1 h. To simulate a selenium-captured calcium filter, CaSeO_3 was synthesized by a precipitation method using sodium selenite (Na_2SeO_3) and calcium nitrate ($\text{Ca}(\text{NO}_3)_2$) solutions. The glass samples were analyzed by an X-ray diffractometer (XRD). Retention of Se in tellurite glasses was analyzed by an X-ray fluorescence spectrometer (XRF) and inductively coupled plasma (ICP). The chemical durability of tellurite glass was evaluated through the PCT method.

Keywords: Selenium, Vitrification, Glass, Chemical durability, Wasteforms

A Study on Suitability for the Vitrification of Radioactive Asbestos Waste in Kori Unit1 Decommissioning

Hwa-jun Yeom¹, Chang-Lak Kim^{1,*}, Chang-Sig Kong¹, and In-Sun Jang²

¹Kepeco International Nuclear Graduate School (KINGS)

²Dawonsys

*clkim@kings.ac.kr

In operating or permanently shut down nuclear power plants which were built between 1970s and 1990s, asbestos was widely used for ceiling materials, wall materials, and gaskets. Furthermore, it was mainly treated as a heat-resistant material like insulation. In Kori Unit 1, radioactive asbestos was replaced or removed through maintenance and repair in the containment building during the operation period of about 40 years, but radioactive asbestos still remains that need to be partially dismantled. Generally, it is more difficult to handle because it belongs to two different waste categories, radioactive waste and hazardous waste. In addition, the risk increases further due to radioactivity with the asbestos hazards itself. Therefore, it is very important to accurately determine the amount of radioactive asbestos waste and to evaluate the treatment method and disposal reduction rate before the decommissioning is started. According to the Korean Waste Management Act, three methods are recommended for the asbestos (hazardous waste) treatment: landfill, solidification, and high-temperature melting. Landfill is commonly used in Korea and the United States while high-temperature melting and solidification are additionally recommended only in Korea. Considering the situation in Korea, landfill is not appropriate due to the limitations of landfill capacity and potential risks (hazards still remain). Therefore, the other two methods can be considered sufficiently in terms of safety, detoxification, and reduction rate. This paper evaluates the amount of radioactive asbestos waste at Kori Unit 1 based on the actual asbestos building material data (as of February 2022) of the Asbestos Management Comprehensive Information Network. Vitrification is considered as a sufficient alternative for treating radioactive asbestos waste. And, it is checked whether the vitrified waste through the high-temperature melting method, plasma torch, meets the requirements such as detoxification, compressive strength and leachability for storage and disposal stability. It is expected to be useful to prepare a radioactive mixed waste management standard and to reduce the disposal cost through the reduction of final waste.

Acknowledgements

This study is supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Mixed Waste, Asbestos, Radioactive, Vitrification, Decommissioning

Reduction and Recycling of Nuclear Dismantled Concrete Waste

Jin Ryu¹, Jun Ki An¹, Gun Hee Kim¹, Ji Eun Shin¹, and Won-Seok Kim^{1,2,*}

¹NCsquare Co.

²Pohang University of Science and Technology (POSTECH)

*wkim23@ncsquare.co.kr

The number of dismantled nuclear facilities is increasing globally. Dismantling of nuclear facilities generates large amount of waste such as concrete, soil, and metal. Concrete waste accounts for 70% of the total amount of waste. Since hundreds of thousands of tons of concrete waste generated, securing technology of reduction and recycling of waste is emerging as a very important issue.

The objective of this study is to synthesize geopolymer using inorganic materials from cement fine powder in concrete waste. Dismantled concrete waste contains a large amount of calcium silicate hydrate(C-S-H), Ca(OH)_2 , SiO_2 , etc., which is an inorganic material required for the synthesis of geopolymer. SiO_2 affects the compressive strength of the geopolymer and Ca(OH)_2 affects the curing rate. A high concentration of alkali solution is used as an alkali activator, and alkali activator is necessary for the polymerization reaction of metakaolinite.

The experiment consists of three steps. The first step is to react with concrete waste and hydrochloric acid to extract ions. In the solid after filtration, SiO_2 and Al_2O_3 are composed of 84.10%. It can be used instead of commercial SiO_2 required for the synthesis of geopolymer. The second step is to add NaOH up to pH 10, impurities can be removed to extract Ca(OH)_2 with high purity. The final step is to add NaOH up to pH 13, and Ca(OH)_2 extraction. The alkali solution generated after the last reaction can be recycled into an alkali activator during the synthesis of the geopolymer. If dismantled concrete waste is recycled during geopolymer synthesized, the volume reduction rate of dismantled concrete waste is more than 50%.

If you put the radioactive waste in the recycled solidification materials synthesis from concrete waste by dismantling of nuclear facilities, it is possible to reduce the amount of waste generated and disposal costs.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020152030 0070).

Keywords: Recycling Calcium Hydroxide, Reduction of Concrete Waste, Geopolymer, Nuclear Power Plants Dismantling

~~International Case Study on Economic Estimation for Thermal Plasma Gasification~~

~~InJun Lee^{1,*}, HyungJun Kim¹, and Seoyong Choi²~~

~~¹KEPCO Korea Plant Service & Engineering (KEPCO-KPS)~~

~~²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)~~

~~*injun@kps.co.kr~~

Withdrawal

Removal of Uranium Ions From Sulfate-Based Acidic Soil Washing Effluent Using Ion-Exchange Resins

Hyun-Kyu Lee*, Seeun Chang, Wooshin Park, Sungbin Park, and Hongrae Jeon

Korea Atomic Energy Research Institute (KAERI)

*hklee@kaeri.re.kr

This study aimed to remove uranium (U(VI)) ions from sulfate-based acidic soil-washing effluent using the ion-exchange method. For effective ion exchange of U(VI) ions under acidic conditions, one chelate resin (Purolite S950) stable under low pH conditions and two anion-exchange resins (Ambersep 400 SO₄ and 920U SO₄) used in sulfuric acid leaching systems were selected. The exchange performance of the three selected ion-exchange resins for U(VI) ions was evaluated under various experimental conditions, including ion-exchange resin dosages, pH conditions, reaction times, and reaction temperatures. U(VI) ion exchange was consistent with the Langmuir model and followed pseudo-second-order kinetics. Thermodynamic experiments revealed that the U(VI) ion exchange by the ion-exchange resins is an endothermic and spontaneous process. On the other hand, U(VI) ions were effectively desorbed from the ion-exchange resins using 0.5 M H₂SO₄ or Na₂CO₃ solution. Overall, on the basis of the results of the present study, we propose that Purolite S950, Ambersep 400 SO₄, and Ambersep 920U SO₄ are ion-exchange resins that can be practically applied to effectively remove U(VI) ions from sulfate-based acidic soil-washing effluents.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 521220-22).

Keywords: Uranium, Ion Exchange, Sulfuric Acid, Soil Washing, Effluent Treatment

Introduction on the 3.5th Generation Plasma Torch Melting Facility

Seoyong Choi*, Jeongsu Jeong, and Sunghoon Hong

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*seoyong.choi@khnp.co.kr

Untreated waste is temporarily stored on the site of the nuclear power plant. In some nuclear power plants, saturation period of temporary storage waste is less than 10 years away. As untreated waste continues to be generated in nuclear power plants, it could also affect management of operations. Accordingly, CRI is developing the 3.5 generation plasma torch melting facility for waste treatment.

The 3.5th generation plasma torch melting facility consists of melter, plasma torch, waste supply device, exhaust gas treatment facility, power supply, etc. Melter is composed of melting chamber for melting control and pyrolysis chamber for waste pretreatment, and dam-type discharge device is adopted to overflow the melt. Plasma torch is hollow type with reversed discharge, has a rating of megawatt class, and has two gas supply lines. It can be used in transfer mode, non-transfer mode and mixed mode. There are three types of device for waste supply. The first is a drum pusher for injecting 200 L drums, the second is a screw-type waste supply and hopper for injecting solid waste, and the third is a nozzle-type waste supply device for injecting liquid waste. Exhaust gas treatment facility was equipped with post combustion chamber, off-gas cooler, high-temperature filter, HEPA filter, reheater, scrubber, ID fan and etc. Power supply of plasma torch operation is designed with a capacity of 1.5 megawatt (Maximum) and consists of channels A and B. Transfer mode, non-transfer mode and mixing mode of plasma torch may be selected through the control of PLC.

This paper introduces the composition and function of the 3.5th generation plasma torch melting facility of CRI. In order to solve the problems arising through the operation of the 3rd generation plasma torch melting facility, an optimization plan is applied.

Keywords: Plasma Torch, Untreated Waste, Waste Supply Device, Waste Treatment

Clearance Level Radioactive Waste Treatment for Reuse

YuJeong Choi*, Yoondo Oh, and Tack-Jin Kim

Korea Atomic Energy Research Institute (KAERI)

*choiyj@kaeri.re.kr

Following a radioactive waste criterion and clearance level radioactive waste Act Article 2.

“The radioactive wastes confirmed by the Commission as having concentration by nuclide not exceeding the value determined by the Commission through incineration, reclamation, recycling, etc”.

The combustible clearance level radioactive wastes like lumbers are incinerated and non-combustible wastes like concreted are buried. The metals clearance level radioactive wastes are recycled after being re-molded. However, the clearance level radioactive waste with keeping its original forms is not common.

Due to the nature of KAERI, the equipment are brought into the radiation-controlled zone for experiments. Those equipment are conservatively considered contaminated and categorized with radioactive waste following nuclear safety acts. In this case, the spectroscopy device which is clearance level radioactive waste is self-disposed for use in non-controlled areas.

The 4 devices are composed of 3 gamma-ray spectroscopy and 1 alpha, beta counting system. Those devices were used for clearance level radioactive waste's radioisotope analysis in Radioactive Waste Form Test Facility which is used in a separated room for analysis. This room will be released in non-radiation controlled area, therefore those devices will be moved to non-controlled area and keep using. Last April self-disposal was reported to the regulatory body and got acceptance last May. Those devices were moved to non-controlled area last July.

This case will be good example for reuse equipment which stop using in radiation controlled area but can keep used.

Keywords: Radioactive Waste Treatment, Clearance Level, Reuse, Recycle

A Study on the Development of the Evaluation Module for Decommissioning of Heavy Water Reactors

Hyun Young Shin, Kyu Tae Park, Sung Ryul Kim, Jung Min Oh, Ba Ro Lee, Ji Ung Kim, Hye Jin Kim,

So Yun Jeong, and Wook Jae Yoo*

Orbitech

*hyshin@orbitech.co.kr

With the aging of nuclear power plants (NPPs) in 37 countries around the world, 207 out of 437 NPPs have been permanently shutdown as of August 2022 according to the IAEA. In Korea, the decommissioning of NPPs is emerging as a challenge due to the permanent shutdown of Kori Unit 1 and Wolsong Unit 1. However, there are no cases of decommissioning activities for Heavy Water Reactor (HWR) such as Wolsong Unit 1 although most of the decommissioning technologies for Light Water Reactor (LWR) such as Kori Unit 1 have been developed and there are cases of overseas decommissioning activities.

This study shows the development of a decommissioning waste amount/cost/process linkage program for decommissioning Pressurized Heavy Water Reactor (PHWR), i.e. CANDU NPPs. The proposed program is an integrated management program that can derive optimal processes from an economic and safety perspective when decommissioning PHWR based on 3D modeling of the structures and digital mock-up system that links the characteristic data of PHWR, equipment and construction methods. This program can be used to simulate the nuclear decommissioning activities in a virtual space in three dimensions, and to evaluate the decommissioning operation characteristics, waste amount, cost, and exposure dose to worker.

In order to verify the results, our methods for calculating optimal decommissioning quantity, which are closely related to radiological impact on workers and cost reduction during decommissioning, were compared with the methods of the foreign specialized institution (NAGRA). The optimal decommissioning quantity can be calculated by classifying the radioactivity level through MCNP modeling of waste, investigating domestic disposal containers, and selecting cutting sizes, so that costs can be reduced according to the final disposal waste reduction. As the target waste to be decommissioning for comparative study with NAGRA, the calandria in PHWR was modeled using MCNP. For packaging waste container, NAGRA selected three (P2A, P3, MOSAIK), and we selected two (P2A, P3) and compared them.

It is intended to develop an integrated management program to derive the optimal process for decommissioning PHWR by linking the optimal decommissioning quantity calculation methodology with the detailed studies on exposure dose to worker, decommissioning order, difficulty of work, and cost evaluation. As a result, it is considered that it can be used not only for PHWR but also for other types of NPPs decommissioning in the future to derive optimal results such as worker safety and cost reduction.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 1810300130 and No. 20203210100370).

Keywords: Decommissioning, Heavy Water Reactor, Waste, Evaluation Module, Integrated Management Program

Analysis on US NRC Technical Position for Concentration Averaging of Radioactive Waste

Jiyoung Yi* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*sunsetrise7@naver.com

Radioactive waste is classified into Intermediate level, low level, and very low potential based on the amount of radioactivity per unit gram, that is, the concentration limit.

This method of classifying radioactivity per unit weight is not a problem if all packaged wastes are homogeneous. However, the reality is that not all waste is homogeneous. Relative hotspots may exist.

Also, when several items are mixed, if one item has a relatively higher concentration than other items, it can become a relative hotspot.

In Korea, even if all nuclides in a single radioactive waste package satisfy the low level concentration limit, if even one nuclide exceeds the concentration limit, the radioactive waste package becomes the intermediate level.

In case of the United States, the US NRC provides regulations related to obtaining license as well as presents the technical position on the average waste concentration called Concentration Averaging and Encapsulation Branch Technical Position (CA BTP).

CA BTP classifies waste into four types : Blendable Waste, Encapsulated items, Single Discrete Items, and Mixture of Discrete Items, and presents each approach to concentration averaging. In general, this is a method that suggests an acceptable ratio in case of the waste, which relatively high concentration waste is mixed.

In order to apply this in Korea, we compare the classification standards for low and Intermediate-level waste in Korea and the United States, related laws and backgrounds, and the application methods of CA BTP.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP Project No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Concentration Averaging, Radioactive Waste, Discrete Item, Blendable Waste, CA BTP

Study on Recycling Strong Acidic Wastewater Containing a Radionuclide for the Radioactive Waste Reduction

Hee-Chul Eun^{1,2,*}, Na-On Chang¹, Tack-Jin Kim¹, and Woo-Shin Park¹

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*ehc2004@kaeri.re.kr

Strong acidic wastewater containing a radionuclide is generated from the decontamination of radioactively contaminated wastes or equipment. This wastewater is generally treated through a precipitation process using an alkali (alkali earth) hydroxides. In this precipitation process, a significant amount of alkali (alkali earth) sulfates are generated, which is the reason for the increase in the radioactive waste generation.

In this study, a method for separating only radionuclides and metal ions from the wastewater was evaluated. For this reason, precipitation behaviors of radionuclides and metal ions by hydrazine injections were investigated using equilibrium calculations. In addition, behaviors of hydrazine decomposition after removal of radionuclides and metal ions were analyzed for recycling the wastewater.

Keywords: Strong Acidic Wastewater, Radionuclide, Hydrazine, Recycling

A 100 kW Class Transferred Type Plasma Torch for Radioactive Waste Melting Applications

Darian Figuera-Michal, Dong-Hyun Lee, Nam-Ki Lee, Shi-Young Yang, and Jun-Ho Seo*

JeonBuk National University (JBNU)

*jhseo@jbnu.ac.kr

We developed a 100 kW Class Transferred Type Plasma Torch applicable for melting of non-combustible metal wastes. By employing reverse polarity discharge structures for hollow electrode plasma torch, a transferred type arc plasma was generated stably with long arc length higher than 10 cm at the arc currents of ~ 400 A and gas (N_2) flow rate of ~ 50 lpm. High arc currents and high arc voltages caused by the increased arc length could input high power level of ~ 100 kW to the non-combustible metal wastes, enabling quick melting. In addition, relatively long arc length and low gas flow rates can help reduce the deposition of melted materials on the exit surface of the torch. Thanks to these features, the developed plasma torch is expected to be suitable for small-scaled and portable melting system.

Keywords: Plasma, Transferred Type Torch, Melting, Radioactive Waste, Non-combustible

A Conceptual Study on Radioactive Waste Solidification Treatment Using Plasma Torch Melter

Jihyun Kim^{1,*}, HyungJun Kim¹, and Seoyong Choi²

¹KEPCO Korea Plant Service & Engineering (KEPCO KPS)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*safetyji@kps.co.kr

KHNP-CRI has developed Mega-Watt Class PTM (Plasma Torch Melter) for the purpose of reducing the volume of radioactive waste and immobilizing or solidifying radioactive materials. About 1 MW PTM is a treatment technology that operates a plasma torch and puts drum-shaped waste into a melter and radioactive waste in the form of slag is discharged into a waste container. Since only the overflowing slag is discharged from the melter, the discharge is intermittent. Therefore, solidification occurs in the process of discharging the melt. It is difficult to accumulate evenly in the waste container, and there is also an empty space. Solid radioactive waste must be disposed of to meet the acceptance criteria for radioactive waste. Plasma-treated solid waste raised concerns about the requirements. The waste solidification output in a slag container gave us some concerns for the waste package's solidification and encapsulation requirements. The plasma-treated solid waste process to meet the acceptance criteria will be cost and need time consuming. Thus, a induction heating will be introduced to meet solidification requirements and test criteria of the solidification waste for the waste package disposal.

Keywords: Plasma Torch Melter (PTM), Radioactive Waste Disposal, Solid Radioactive Waste, Induction Heating

Development of Procedures for Compression and Packaging of Spent Filter

Narae Lee¹, Moonoh Kim^{1,*}, Ji Soo Yoon², and Ji Hoon Lee²

¹KONES Corp.

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*mokim@kones21.com

In this study, the process of compressing/packaging the spent filters of Kori Unit 1, which was conceptually presented in the previous study, is advanced so that disposal suitability for each step can be secure efficiently. In particular, the differences between the previous study and this study are that the disposable filters are screened using an In-Situ Object Counting System (ISOCS), and the method of collecting representative samples for development of scaling factor is specified. The process of compressing/packaging the spent filters consists of 7 stages as follows. 1) Collecting: The spent filters temporarily stored in the filter room are collected by dose and type remotely using a robot system to minimize the radiation exposure of workers according to a pre-established packaging plan. 2) Screening: The gamma activity concentration of the spent filters received by the robot system is measured by ISOCS. The spent filters below the low-level waste concentration limit and the surface dose are transferred into the compression system, while the others are returned in the filter room again. 3) Sampling: The external perforator drilling/cutting the filter was developed for sampling required for the new scaling factors. Since the sampling is collected remotely, the risk of exposure to workers can be reduced. The newly developed scaling factor will be used to verify the disposal suitability of the packages. 4) Compression: According to the pre-established plan, the spent filter collected by dose and type, is supplied to the compression system considering the dose and radionuclide inventory. Whether to additionally store the compressed filter in the drum is determined by checking the accumulated dose. 5) Immobilization: Immobilization with a safety material is necessary when inhomogeneous wastes, like spent filters, have the total radionuclide concentration with a half-life of more than 20 years is 74,000 Bq/g or more and for filling rate or non-dispersible treatment of particulates. 6) Packaging and Analysis: Waste information is labelled onto the package after the measurements of surface dose rate and surface contamination. Finally, using the drum assay system, the gamma radionuclide concentration is measured to identify at least 95% of the total radioactivity concentration of the package. 7) Temporary Storage and Delivery: The packages are moved to temporary storage in the plant prior to disposal. After establishing the plan for delivery and applying for a takeover request to KORAD, if the acceptance inspection is passed, the packages are transported to the disposal facility.

Keywords: Spent filters, Disposal suitability, Compression, Packaging

Performance Test of A 100 kW Class Mobile Plasma Melting System for Non-combustible Wastes

Darian Figuera-Michal¹, Dong-Hyun Lee¹, Nam-Ki Lee¹, Sang-Min Jeong², and Jun-Ho Seo^{1,*}

¹JeonBuk National University (JBNU)

²ORION EnC

*jhseo@jbnu.ac.kr

In this work, we report the basic performance of a 100 kW class mobile plasma melting system consisting of two 24-ft commercial containers, each in charge of the plasma utilities and melting process. In this system, a 100 kW class transferred type plasma torch has been installed together with a crucible having an inner volume of 2,884 cm³. Filling the inner volume of the crucible with the simulated metal waste, such as bolts and nuts, melting tests have been carried out for 5 min by varying plasma input power from 50 kW to 100 kW. By measuring the volume of metal waste before and after melting test, then, the volume reduction rates were estimated and discussed.

Keywords: Plasma Melting, Mobile, Non-combustible Wastes, Performance Test, Volume Reduction Rates

Evaluation of the Mobility of Glass Fibers in the Environment of the Gyeongju Radioactive Waste Repository

Sang June Park^{1,*}, Moonoh Kim², Hyun Woo Song², Hang-Rae Cho¹, and Suil Bang¹

¹Korea Hydro & Nuclear Power (KHNP)

²KONES Corp.

*sjpark21@khnp.co.kr

Glass wool, the primary material of insulation, is composed of glass fibers and is used to insulate the temperature of steam generators and pipes in nuclear power plants. Glass fiber is widely adopted as a substitute for asbestos classified as a carcinogen. The insulations used in nuclear power plants are classified as radioactive waste and most of the insulation is Very Low-Level Waste (VLLW). It is packaged in a 200 L drum the same as a Dry Active Waste (DAW). In the case of the insulations, it is packaged in a vinyl bag and then charged into the drum for securing additional safety because of the fine particle size of the fiberglass. A safety assessment of the disposal facility should be considered to dispose of radioactive waste. As a result of analyzing overseas Waste Acceptance Criteria (WAC), there is no case that has a separate limitation for glass fiber. Also, in order to confirm that glass fibers can be treated in the same manner as DAW, research related to the diffusion of glass fibers into the environment was conducted in this paper. It was confirmed that the glass fiber was precipitated due to the low flow velocity of groundwater in the Gyeongju radioactive waste repository and did not spread to the surrounding environment due to the effect of the engineering barrier. Therefore, the glass fiber has no special issue and can be treated in the same way as a DAW. In addition, it can be disposed of in the disposal facility by securing sufficient radiological safety as VLLW.

Keywords: Glass Fiber, Radioactive Waste Repository, Safety Assessment, Mobility

Proposal for the Composition of the Waste Profile (Draft) for Large Package of Decommissioning Waste

Hoseog Dho*, Hyungoo Kang, Yeseul Cho, and Chunhyung Cho

Korea Radioactive Waste Agency (KORAD)

*ehghtjr@korad.or.kr

The decommissioning waste generated during the dismantling of a nuclear power plant has various types and radioactivity levels and is characterized by being generated in a large amount in a short time. For the safe and efficient management of decommissioning waste with these characteristics, the Korea Radioactive waste Agency (KORAD) is developing a large container for decommissioning waste. And the Waste Certification Program (WCP) requirement was developed for the development of a Waste certification program for nuclear power plant operators that can prove whether the transfer/acceptance criteria are satisfied at the step of generation of decommissioning waste packages. The radioactive waste profile, which is a waste quality certification document required by the WCP requirements, allows the KORAD to confirm that the disposition suitability evaluation was performed for each process of decommissioning waste and radiological characteristic data were evaluated appropriately. Therefore in this study, in order to propose a draft of a radioactive waste profile for large packages of decommissioning waste, overseas cases and the draft radioactive waste profile of the WCP requirements was analyzed. In addition, it was attempted to increase the utility of the derived waste profile by clearly suggesting the treatment methods for each waste stream considering the physical and radioactive characteristics evaluation methods of large decommissioning waste packages. The proposed large decommissioning waste profile can be used in the future development of a nuclear power plant operator's decommissioning waste certification program, as well as KORAD's a disposal facility safety evaluation and improvement of the waste tracking management system (WTS).

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP Project No.20211510100040) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Decommissioning Waste, Large Radioactive Waste Package, Waste Characterization Profile, Waste Certification Program (WCP)

Design of Three Phase AC (Alternative Current) Arc Plasma System For Treatment of Radioactive Slurry Wastes

Dong-Hyun Lee, Darian Figuera-Michal, Hae-Won Park, Sang-Yun Baek, and Jun-Ho Seo*

JeonBuk National University (JBNU)

*jhseo@jbnu.ac.kr

In the present work, a three-phase AC arc plasma torch system is proposed to separate inorganic radioactive materials from the organic liquid waste. For this purpose, first, assuming the resistance of arc plasma ranges between 0.1 and 0.2 ohm, we designed a three-phase AC arc plasma power supply with the power level of 20 kW. Then, a three phase arc plasma torch consisting of three carbon rods with the diameter of 20 mm was designed and mounted on a cylindrical combustion chamber with the inner diameter of 150 mm. Detail design and basic performance of the plasma system were presented and discussed for application to the treatment of radioactive slurry wastes.

Keywords: AC Arc Plasma, Slurry Wastes, Combustion, Organic Waste

Evaluation of Seawater Intrusion Characteristics Using the Long-range GPR Systems for Coastal Line at the Site of LILW Disposal Facilities

Seung Gyu Hyun^{1,*}, Hoseon Choi¹, and Youngchul Yu²

¹Korea Institute of Nuclear Safety (KINS)

²KOTAM

*mgodo@kins.re.kr

During and after the construction of LILW disposal facilities, the decrease of groundwater head potential has been monitored. In addition, an increase of the electrical conductivity (EC) has been observed in several monitoring wells installed along the coastal coastline. Monitoring activity for groundwater head potential and hydrogeochemical properties is important to reduce the uncertainty in the evaluation of groundwater flow characteristics. However, the data observed in the monitoring wells are spatial point data, so there is a limit to the dimension.

Several researchers evaluated groundwater head potential changes and seawater intrusion (SWI) potential for disposal sites using groundwater flow modeling. In case of groundwater flow modeling results for SWI, there is a spatial limit in directly comparing the EC observed in the monitoring wells with the modeling results.

In a recent study, it was confirmed that the response of the long-range ground penetrating radar (GPR) system was severely attenuated in the presence of saline groundwater. In order to reduce the spatial constraint of the groundwater monitoring wells for SWI, the characteristics of SWI within the disposal facility site by using the the results of a recent study of the long-range GPR system were investigated and evaluated in this study.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1805020).

Keywords: LILW, Groundwater Flow, Seawater Intrusion, Monitorig, GPR

The Evaluation of Tritium Behavior by Replacing the Reflector Ion Exchanger Resin With HANARO

Hoon-Jo Cho*, Mun-Jo Choi, Kwang-Young Lee, Hee-Gon Kim, and Ho-Young Choi

Korea Atomic Energy Research Institute (KAERI)

*chj1869@kaeri.re.kr

HANARO, a multi-purpose research reactor, uses a reflector as heavy water to obtain high neutron flux. Therefore, two ion exchangers were installed to manage the heavy water quality of the reflector system. The operator of HANARO manages it according to the limit value (Conductivity: less than 0.5 mS/m, pH: 5.5~6.5), and if the limit value is not satisfied, the resin must be replaced. The reflector system is in the enclosed structure and it is designed to delay the release of tritium to the outside.

Tritium is produced by a nuclear reaction between neutrons and deuterium. Tritium is inhaled into the body in the form of water or vapor, which is likely to cause internal exposure problem. In addition, since tritium spreads to other regions, thorough management is required. Therefore, HANARO measures and manages tritium in Rx and RCI using the bubbler collection method.

In this paper, the change in the behavior of tritium due to the replacement of the reflector ion exchanger resin was analyzed. Due to the increase in conductivity of the reflector, the ion exchanger resin was replaced on March 3, 2022. Therefore, the concentration of tritium was measured to be about 5 times higher than usual. It did not exceed the emission limit, and the concentration values of tritium is stably managed by constant monitoring and analysis.

Keywords: HANARO, Tritium, Reflector, Bubbler Collection Method

Characteristics of Waste-form Containing Chemical Decontamination Waste Immobilized by Low-temperature Sintering

Naon Chang¹, Hee-Chul Eun^{1,2,*}, and Won-Hyuk Jang¹

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*eun2004@kaeri.re.kr

In a nuclear power plant, the activated corrosion products are deposited on the reactor coolant system. The activated corrosion products must be removed to reduce the radiation exposure to workers before maintaining or decommissioning of the nuclear power plant. In order to remove the activated duplex oxide layer generated on the reactor coolant system in the pressurized water reactor (PWR), the Cyclic SP (Sulfuric acid/Permanganate)-HyBRID (Hydrazine Based Reductive metal Ion Decontamination) process developed by KAERI (Korea Atomic Energy Research Institute) can be used. After applying the Cyclic SP-HyBRID process, a sulfate-rich waste powder containing the radionuclide is generated, and the radioactive powder has to be stabilized for final disposal. In the previous study, it was confirmed that the low-temperature sintering method can be applied to immobilize the sulfate-rich waste powder. Thus, immobilization of the Cyclic SP-HyBRID process waste powder was carried out by the low-temperature sintering method using a low melting point glass, and the physicochemical and radiological characteristics of a waste form were evaluated in this study. As a result, the compressive strength of the waste form increased with increasing sintering temperature and sintering time. It is considered that the result was caused by the difference in the band gap between the bismuth borate and zinc borate, which are the products during the sintering process. It was verified that the physical stability was maintained after the 10^7 Gy of irradiation test. In addition, it was confirmed that the radioactive metal hydroxides contained in the waste powder converted to metal oxide forms, which have the lower solubility, at the sintering temperature. Finally, the waste form was evaluated as a low-level radioactive waste from the concentration of radionuclides contained in the waste form.

Acknowledgements

This work was supported by a research grant from the Korea Atomic Energy Research Institute (KAERI) (Grant No. 521230-22).

Keywords: Radioactive Waste, Immobilization, Waste Form, Sintering, Radionuclide

The Effect of ISA on the Sorption and the Solubility of Nickel

Soojeong Ji and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongum@postech.ac.kr

Radioactive nickel (Ni^{59} and Ni^{63}) is a major radionuclide that needs to be determined for quantifying the total radioactivity in radioactive waste disposal repository. Also, radioactive waste containing organic wastes, such as cotton and tissue can be decomposed to produce the Isosaccharinic acid (ISA) in a disposal facility. The presence of ISA in the disposal facility could increase the mobility of radionuclides. Therefore, it is necessary to confirm the mobility of Ni with the presence of ISA in the repository. This study investigated the effect of ISA on the sorption and the solubility of Ni in synthesized groundwater. The sorption test was conducted in different time intervals with Ni and ISA. Nickel nitrate hexahydrate and $\text{Ca}(\text{ISA})_2$ were used after purchase. Granite was used as the solid medium to simulate the major rock type of the repository. Ni and ISA solution with the medium were mixed using a platform shaker for 6 days. After 6 days, the solid parts were separated by centrifugation and additional syringe filters, and the supernatant was analyzed for Ni and ISA concentration using ICP-MS and IC, respectively. The solubility experiments were conducted at different temperatures (20, 40, and 80°C). Nickel hydroxide was used as the solubility limiting solid phase. To balance the ionic strength and confirm the effect of ISA on Ni solubility, 0.01 M of CaCl_2 solution was prepared in a sample without ISA, and 0.01 M of $\text{Ca}(\text{ISA})_2$ solution was prepared in a sample with ISA. In solubility tests, the solution was also analyzed by ICP-MS and IC for Ni and ISA, respectively. The concentration of Ni was found to increase with ISA compared to Ni concentration without ISA. The concentration of ISA was not changed during the solubility test periods. For solubility tests, the concentration of Ni also increased according to the increase in temperature. The solid phase was characterized by XRD, FT-IR, and SEM-EDS. Based on the results of this study, the presence and effect of ISA on radioactive Ni mobility should be carefully investigated to secure the long-term safety assessment for the low and intermediate-level waste repository.

Acknowledgements

This research was supported by Ministry of Science and ICT, Korea (2021M2E1A1085203).

Keywords: Radionuclides, Sorption, Solubility, Nickel ISA

Simulation of Liquid Waste From Decontamination of Small- and Medium-sized Metal Wastes of NPP Decommissioning and Design Conditioning and Treatment Process

Seong Ho Yoo*, Ki Seock Cho, Kwang Nam Kim, and Ho Yeon Yang

Korea Environment Testing and Analysis Laboratory

*shyoo@keta.re.kr

Organic waste generated by small and medium-sized (S&M-sized) metal decontamination in NPP decommissioning. To lower the concentration of these organic substances for a level acceptable at the disposal site, the project of “Development of Treatment Process of Organic Decontamination Liquid Wastes from Decommissioning of Nuclear Power Plants” is being carried out. The conditioning and treatment process of organic liquid waste was designed. Also, the literature was investigated to make simulated organic liquid waste, and the composition of these waste was analyzed and compared.

As the decontamination agent, organic acids such as EDTA, oxalic acid, citric acid are used. The sum of the concentrations of these organic materials was set to a maximum value of 1,000 ppm. The major metal ions of the decontamination liquid waste estimated are ^{59}Fe , ^{51}Cr , ^{54}Mn , ^{63}Ni , and the concentrations are respectively 527, 163, 161, 159 ppm. Additional major metal ions are ^{60}Co , ^{58}Co , ^{137}Cs . ^{58}Co is replaced by ^{60}Co because it has the same chemical properties as ^{60}Co . Unlike the HLW, the contamination level of S&M-sized metal in primary system was quite low, so ^{60}Co is set to 2,000 Bq/g. Considering the contribution of fission and gamma ray dose constant, ^{137}Cs was estimated to 360 Bq/g. Also, suspended solids of decontamination liquid waste were set at 500 ppm.

Under these assumptions, the simulated organic liquid waste was made, and then organic substances and metal ions were analyzed with TOC analyzer and ICP-OES. The TOC analysis value was expected to 392 ppm in consideration of the equivalent organic quantity. the test result was 302 ppm. Some of organics appears to have been decomposed by acid. The values of metal ions (Fe^{3+} , Cr^{3+} , Mn^{2+} , Ni^{2+}) analyzed by ICP-OES are 139, 4, 152, 158 ppm, respectively. A large amount of Cr^{3+} and Fe^{3+} were expected to exist as ions, but they existed in the form of suspended solid. Mn^{2+} and Ni^{2+} came out similar to the expected values.

The designed conditioning and treatment process is largely divided into pretreatment, conditioning, and decomposition processes. After collecting in the primary liquid waste storage tank, large particulate impurities and suspensions are removed through a pretreatment process. In the conditioning process, treated liquid waste passes through UF/RO membrane system, and pure water is discharged to the environment after monitoring. Concentrated water is decomposed in the electrochemical catalyst decomposition process, then this water secondarily passes through the RO membrane system and then discharged to the environment after monitoring. Through an additional experiment, the conditioning and treatment process will be verified.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 1520300130).

Keywords: NPP Decommissioning, Organic Decontamination Liquid Wastes, Conditioning, Treatment Process

Considerations for Designing Solidification Devices of Dispersible Radioactive Waste

Dong-Sik Jin*, Yong-Ho Hong, Hui-Gyeong Kim, Sang-Soo Kwak, and Jae-Geun Lee

Nuclear Environment Technology Co., Ltd.

*dsjin1064@netec.co.kr

Encapsulation using cement as a solidification method for disposal of radioactive waste is most commonly used in most advanced countries in the nuclear technology to date due to its advantages such as low material cost and accumulated technology. However, in case of cement solidification, since moisture or hydroxyl group in cement is decomposed by radioactivity, it may happen that gas is generated, structural stability is weakened, and leachability is increased due to low chemical durability.

Therefore, the various new solidification methods are being developed to replace it. As one of these alternative technologies, for dispersible metal compounds generated through the incineration replacement process, the study on engineering element technology for powder metallurgy is under way, which overcomes the interference problem between mechanical elements and media that may occur during the process such as the homogeneous mixing process of the target powder substance and additives used in the powder metallurgy concept-based sintering process for the solidification of the final glass composite material (GCM), the process of creating a compressed molded body using a specific mold, the process of final sintering treatment.

The solidification process of dispersible radioactive waste can be largely divided into pre-treatment stage, molding stage, and sintering stage, and the characteristics of the final radioactive waste solidification material can vary depending on the solidification treatment characteristics of each stage. In relation with these characteristics, the matters to be considered when designing device for each stage to solidify dispersible radioactive waste (property of super-mixing device for homogenized powder formation, structural geometry and pressure condition of molding device for production of compressed molded body, temperature and operation characteristics of sintering device for final glass composite material (GCM), etc.) are drawn out. It is expected that the solidification device design reflecting these considerations will meet all disposal conditions of radioactive waste material, such as compressive strength and leaching characteristics of solidified radioactive waste material, and create a uniformized solidification of radioactive waste material.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (RS-2022-00155257).

Keywords: Dispersible Radioactive Waste, Glass Composite Material (GCM), Disposal Conditions, Solidification Device

A Conceptual Design for the Development of a Gripper Device for Lifting Work of Polymer Concrete High Integrity Container (PC-HIC) Packages

Hang-Rae Cho^{1,*}, Hyung-Jin Roh¹, Mi-Hyun Lee², and Sang-June Park¹

¹Korea Hydro & Nuclear Power (KHNP)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hrcho6795@khnp.co.kr

High Integrity Container (HIC) made of polymer concrete was developed for the efficient treatment and safe disposal of radioactive spent resin and concentrate waste in consideration of the disposal requirements of domestic disposal sites. Permission for application of Polymer Concrete High Integrity Container (PC-HIC) to the domestic nuclear power plants (NPPs) has been completed or is under examination by the regulatory agency. In the case of 860 L PC-HIC for very-low-level-waste (VLLW) or low low-level-waste (LLW), the application of four representative NPPs has been approved, and the license for extended application to the rest NPPs is also almost completed. A licensing review is also underway to apply 510 L PC-HIC for intermediate and low-level-waste (ILLW) to representative nuclear power plants. In order to handle and efficiently store and manage PC-HICs and high-dose PC-HIC packages, a gripper device that can be remotely operated and has excellent safety is essential, and the introduction of NPPs is urgent. The conventional gripper device developed by the PC-HIC manufacturer for lifting test to evaluate the structural integrity of PC-HIC requires a rather wide storage interval due to its design features, and does not have a passive safety design to handle heavy materials safely. In addition, work convenience needs to be reinforced for safety management of high radiation work. Therefore, we developed a conceptual design for a gripper device with a new concept to minimize the work space by reflecting on-site opinions on the handling and storage management conditions of radioactive waste in NPPs, and to enhances work safety with the passive safety design by the weight of the package and the function of checking the normal seating of the device and the normal operation of the grip by the detector/indicator, and to greatly improves the work efficiency and convenience with the wireless power supply function by rechargeable battery and the remote control function by camera and wireless monitoring & control system. Through design review by experts in mechanical system, power supply and instrumentation & control fields and further investigations on the usage conditions of PC-HICs, it is planned to facilitate preparations for the application of PC-HIC to domestic NPPs by securing the technical basis for a gripper device that can be used safely and efficiently and seeking ways to introduce it in a timely manner.

Keywords: Gripper Device, Polymer Concrete Container, Lifting Work of PC-HIC, Passive Safety Design

Study on Coincidence Summing Corrections Using LVis Software in Analysis of the Low-level Radioactive Waste

Kyunghun Jung*, Minyoung Jung, Jung Bo Yoo, and Tae-Hong Park

Korea Atomic Energy Research Institute (KAERI)

*kyunghunjung@kaeri.re.kr

According to the ‘Regulations on the Delivery of Low and Medium Level Radioactive Waste’, Notification No. 2021-26 of the Nuclear Safety and Security Commission, a history of radioactive waste and a total amount of radioactivity in a drum are mandatory. At this time, the inventory of radionuclides that make up more than 95% of the total radioactivity contained in the waste drum should be identified, including the radioactivity of H-3, C-14, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Nb-94, Tc-99, I-129, Cs-137, Ce-144, and total alpha.

Among nuclides to be identified, gamma-emitting nuclides are usually analyzed with a gamma ray spectrometer such as HPGe. When a specific gamma-ray is measured with a detector, several types of peaks generated by recombination or scattering of electrons are simultaneously detected in addition to the corresponding gamma-ray in gamma-ray spectroscopy. Among them, the full energy peak efficiency (FEPE) with the total gamma energy is used for equipment calibration.

However, this total energy peak efficiency may not be accurately measured due to the coincidence summing effect. There are two types of coincidence summing: Random and True. The random coincidence summing occurs when two or more gamma particles emitted from multiple nuclides are simultaneously absorbed within the dead time of the detector, and this effect becomes stronger as the counting rate increases. The true coincidence summing is caused by simultaneous absorption of gamma particles emitted by two or more consecutive energy levels transitioning from single nuclide within the dead time of the detector. This effect is independent of the counting rate but affected by the geometry and absolute efficiency of the detector. The FEPE decreases and the peak count of region where the energy of gamma particles is combined increases when the coincidence summing occurs.

At the Radioactive Waste Chemical Analysis Center, KAERI, samples with a dead time of 5% or more are diluted and re-measured in order to reduce the random coincidence summing when evaluating the gamma nuclide inventory of radioactive waste. In addition, a certain distance is placed between the sample and the detector during measurement to reduce the true coincidence summing. In this study, we evaluate the coincidence summing effect in our apparatus for the measurement of radioactive waste samples.

Keywords: HPGe, LVis, Coincidence Summing, Radioactive Waste

Solubility Experiment for the Potential Role of Concrete Admixture on the Radionuclide Mobility

Donghun Pak, Jueun Kim, Kyungwon Kim, and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongum@postech.ac.kr

Polycarboxylic ether-based high-range water reducer (PCE) has been proposed to use due to the operational advantages of reduced water content and increased fluidity of cementitious mixtures. But the concern about using PCE can increase the mobility of radionuclides as well. Nuclear Decommissioning Authority (NDA) showed that the PCE formulations increased radionuclide solubility in free solution. Solubility of U(VI), ^{239}Pu , ^{241}Am with the cementitious materials tested with 3:1 pulverized fuel Ash/Ordinary Portland Cement (PFA:OPC) and 9:1 Ground Granulated Blast Furnace Slag/OPC (GGBS:OPC) with PCE that increased at least one and, in some cases, more than three orders of magnitude (between 10^{-9} and 10^{-4} mol dm $^{-3}$) for these radionuclides in the cement-equilibrated solution. It is possible that the relatively low molecular weight substances present in the PCE cement mixture increase the solubility of radionuclides. In addition, the organic substances that are easily miscible with water can contribute to increase the solubility. In this study, several radionuclides (Nb, Ni, Pd, Zr, and Sn) that may be present in intermediate and low-level waste (LIW) repositories were selected based on the half-life and the estimated dose accordingly, and the solubility tests were conducted with and without PCE in solution. To simulate the field condition of the underground repository, synthetic groundwater was prepared based on the recipe by the KAERI Underground Research Tunnel (KURT) DB-3 GW and used as a solvent. The solubility limiting solid phase (SLSP) of each radionuclide was determined using Geochemist's WorkBench (GWB) model. The selected solid phases are Ni(OH) $_2$, ZrSiO $_4$, Nb $_2$ O $_5$, Pd(metal), and SnO $_2$, respectively, and the solubility experiments were conducted with 1.0wt% of PCE per total weight and 0.5 g / 250 ml of selected radionuclide's SLSP for 90 days at room temperature (25°C). Compared with and without PCE presence in solution, the selected radionuclides also showed an increased solubility with the presence of water reducing agent like PCE. This results can be used to correctly estimate the mobility of target radionuclides with the presence of PCE in repository environments.

Keywords: Water Reducing Agent, Polycarboxylic Ether-based Copolymer, Solubility, Radionuclides, Complexation

Release of Sr From Sr-Coprecipitated Calcite Under Various Environment Conditions

Nara Jeong and Wooyong Um*

Pohang University of Science and Technology (POSTECH)

*wooyongum@postech.ac.kr

Concrete is used as the main engineering barrier in low and intermediate level radioactive waste disposal facilities. As the time passed, the radionuclides stored in repository may contact with groundwater and leak into the ecosystem through the rock media. In this process, the radionuclides can react with calcite via sorption or coprecipitation, because calcite is the major mineral of concrete. Under the various background conditions in repository, frequent dissolution-precipitation reactions can happen. Dissolution of Sr-coprecipitated calcite may be different from that of $\text{SrCO}_{3(s)}$ which can mislead the safety performance of radioactive Sr and the estimate of Sr mobility based on the solubility of $\text{SrCO}_{3(s)}$. Strontium is not only one of the fission products but also emits beta rays with a long half-life almost 29 years. The strontium may be released or retarded by the dissolution-precipitation reactions in repository. In this study, the dissolution of Sr-coprecipitated with respect to calcite was tested in various environment conditions. The Sr-coprecipitated calcite, $(\text{Sr,Ca})\text{CO}_{3(s)}$ was synthesized by coprecipitation method in alkaline condition. The 250 mL of 0.1 M of CaCl_2 solution was mixed with 250 mL of 1.14 mM $\text{SrCl}_2 \cdot 6\text{H}_2\text{O}$ solution. Then, independently prepared 500 mL of 0.1 M Na_2CO_3 solution was mixed with the mixed solution of CaCl_2 and SrCl_2 . The precipitates could be made and they were aged for 3 days at room temperature. Then, the supernatant was separated by the centrifugation and the solid at the bottom was dried in an oven at temperature 80°C . After that, the Sr-coprecipitated calcite powder was washed using the DI water several times and dried again before use. Characterization of solid powder was conducted by XRD and SEM, and the ICP-MS and ICP-AES were used to analyze the concentrations of Ca and Sr. The batch dissolution experiment was conducted with a solid-to-solution ratio of 10 g/L groundwater in polyethylene tubes. The oxidative groundwater was synthesized by simulating the chemical composition of KAERI Underground Research Tunnel (KURT) DB-3 groundwater. Different temperatures and pHs were prepared and tested for the release of Sr and Ca from the coprecipitated $(\text{Sr,Ca})\text{CO}_{3(s)}$ to compare the results with the release of Sr and Ca from $\text{SrCO}_{3(s)}$ and $\text{CaCO}_{3(s)}$, respectively. Such as, these results will be used to provide better understanding of Sr release and mobility in various repository environments.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085202).

Keywords: Strontium, Calcite, Release, Dissolution, Coprecipitation

Sorption of Organic Complexing Agents to Engineering Barriers of a Radioactive Waste Disposal Site

Jinseob Kim, Ahsan Abdul Ghani, Bolam Kim, Youngsu Lim, and Dae Sung Lee*

Kyungpook National University (KNU)

*daesung@knu.ac.kr

Organic complexing agents may affect the mobility of radionuclides at low- and intermediate-level radioactive waste repositories. Especially, isosaccharinic acid (ISA) is the main cellulose degradation product under high pH conditions in cement pore water. ISA can combine with radionuclides and form stable complexes that adversely influence adsorption in the concrete phase, resulting in radionuclides to leach to the near- and far-fields of repositories. This study focuses on investigating the sorption of ISA onto engineered barriers such as concrete, thereby studying adsorption isotherms of ISA on concrete and comparing various isotherm models with the experimental data. The adsorption experiment was conducted in three background solutions, groundwater (adjusted to pH 13 using NaOH), State 1 (artificial cement pore water, pH 13.3), and State 2 (artificial cement pore water, pH 12.5), in a batch system at a temperature of 20°C. Concrete was characterized using BET, Zeta-potential analyzer, XRD, XRF, and SEM-EDS. ISA concentrations were detected using HPLC. The experimental data were best fitted to one-site Langmuir isotherm; On the other hand, either two-site isotherm or Freundlich isotherm couldn't give reasonable fitting to the experimental data. The observed ISA sorption behavior on concrete is crucial for the disposal of radioactive waste because it can significantly lower the concentration of ISA in the pore water. Although one-site Langmuir isotherm might effectively represent the sorption behavior of ISA on concrete, the underlying mechanism is still unknown, and further investigation should be done in the near future.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100370, Development of the evaluation technology for the treatment and disposal of organic complexing agents generated from decommissioning of nuclear power plants). This research was partially supported by the Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (NRF-2018R1A6A1A03024962).

Keywords: Isosaccharinic Acid, Sorption Isotherm, Concrete, Langmuir Isotherm, Radionuclide

Safety Tests of Waste Containers for Packaging of NPP Decommissioning Wastes

Ju-Chan Lee*, Jongmin Lim, Yun Young Yang, Sangsoon Cho, and Jeong-Yong Park

Korea Atomic Energy Research Institute (KAERI)

*sjclee@kaeri.re.kr

Waste containers for packaging, transportation and disposal of NPP (Nuclear Power Plant) decommissioning wastes are being developed. In this study, drop tests were conducted to prove the safety of containers for packaging of the wastes and to verify the reliability of the analysis results by comparing the test and analysis results. The drop height of the waste containers was considered to be 30 mm, which is the maximum lifting speed of a 50 tons crane in the waste treatment facility converted to the drop height. Drop orientation of the containers was considered for bottom-end on drop. The impact acceleration and strain data were obtained to verify the reliability of the analysis results. Before and after the drop tests, measurement of the dose rate and the radiographic testing for concrete wall, and measurement of the wall thickness of steel plate were conducted to evaluate the radiation shielding integrity. Also, measurement of bolt torque, and visual inspection were conducted to evaluate the loss or dispersion of radioactive contents. After the drop tests, the radiation dose rate on the container surface did not increase by more than 20%, and there was no crack in the concrete. In addition, the thickness of the steel plate did not change within the measurement error. Therefore, the radiation shielding integrity of the container was maintained. After the drop tests, the lid bolts were not damaged and there was no loss of pretension in the lid bolts. In addition, there was no loss or dispersion of the contents as a result of visual inspection. In order to prove the reliability of the drop analysis results, safety verifications were performed using the drop test results, and the appropriate conservatism for the analysis results and the validity of the analysis model were confirmed. Therefore, the structural integrity of the waste containers was maintained under the drop test conditions.

Acknowledgements

This work was supported by the KETEP (Korea Institute of Energy Technology Evaluation and Planning) and the MOTIE (Ministry of Trade, Industry and Energy) of the Republic of Korea (No. 20211510100040).

Keywords: Safety Test, Waste Container, Packaging, Decommissioning Waste, Drop Test, Safety Verification

Review on Measurement Method of Level Using Differential Pressure in Nuclear Facility

Yoon-do Oh*, Hyun-myung Jung, Dae-seok Hong, Dong-han Yoo, Hyun-kyu Lee, and Tack-jin Kim

Korea Atomic Energy Research Institute (KAERI)

*ydoh@kaeri.re.kr

There are various types of level gauging method such as using float, differential pressure, hypersonic, displacement and so on. In this study, among them, the method utilizing the differential pressure was reviewed.

The strengths include: the differential pressure type level gauge can measure the level without direct contact of the sensor with media. That is to say, the level can be measured even if the sensor is far away from the tank. And regardless of the size of the tank, the level can be measured if the pneumatic pipes are installed. The weaknesses include: the sensor needs intermedium to recognize the level. The intermedium utilizes a fluid, which is compressed air. It is difficult to handle that compressed air has the properties of a gas. And to make compressed air needs compressor, tank and pneumatic pipes. So if you have many tanks, you need to install exponentially the pneumatic pipes. As well, level measurement range is limited to the points where the pneumatic pipes of the tank is installed. And if a compressed air that supplies to the sensor leaks, uncertainty will increase. A compressed air is colorless and odorless, so it's difficult to pinpoint the leak. Finally, events like cracks and clogging can cause inaccurate measurement.

Rather than using only differential pressure, it is better to use another measurement method according to the situation of the facility.

Keywords: Differential Pressure, Liquid Tank, Level Measurement

Process for Regulatory Clearance of Wood Waste at KAERI

Han Byeol Kang*, Jong Kwang Lee, and Tack-Jin Kim

Korea Atomic Energy Research Institute (KAERI)

*khh80@kaeri.re.kr

In KAERI, Waste storage facility in the radiation management area has stored a large amount of wood waste. The amount of waste is approximately 27,000 kg, it accounts for 17% of the total waste in waste storage facility. Proper disposal of wood waste improves the fire resistance performance, secure storage space and reduce disposal costs. In order to self-disposal of wood waste, it is necessary to satisfy the self-disposal standards stipulated by the domestic Atomic Energy Act and the treatment standards of the Waste Management Act. The main factors of standards are surface contaminant, radionuclide activity and radiation dose effects. To confirm the contamination of wood waste, direct indirect measurement methods and gamma nuclide analysis were performed. To evaluate radiation dose, various computational programs were used. The results of the analysis were satisfied with domestic regulations on the classification and self-disposal of radioactive wastes. Based on this results, KAERI submitted the report on wood waste self-disposal plan to obtain approval. After final approval, wood waste is to be incinerated and incineration ash is to be buried in the designated place. The objective of this study is to provide total procedure of wood waste self-disposal and effective representative sampling method.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 521210-22).

Keywords: Wood Waste, Clearance, Self-disposal

Calculation of Radiation Dose Using Computational Program NRCDOSE

Han Byeol Kang*, Ki-Baek Shin, and Tack-Jin Kim

Korea Atomic Energy Research Institute (KAERI)

*khh80@kaeri.re.kr

In general, dose assessment must be performed to obtain approval for clearance of radioactive waste. If the annual dose criteria through dose evaluation satisfies the clearance condition, radioactive waste can be disposed of. Various programs are used to perform dose assessment. NRCDOSE GASPAR is used as a program to assess the amount of radiation exposed to atmospheric emissions. Program is easy to use and results can be checked immediately after execution. GASPAR requires main input factors by exposure route such as site specifics, source term, special location, block data. Basically, program has default input values but user can easily modify it. The most important factor is that when entering a nuclide, the effect on progeny radionuclides is not automatically calculated. User should consider the dose contribution from progeny radionuclides.

In this study, dose assessment was performed for combustible waste incineration using NRCDOSE GASPAR. And it was confirmed that exposure dose of individuals and groups criteria for clearance regulation.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 521210-22).

Keywords: NRCDOSE, Clearance, Dose Assessment

Development of Integrated Decontamination Equipment to Remove Concrete Surface Contamination

Byeong-Ho Kim^{1,*}, Ki-Hyun Kwon¹, Hyeon-Oh Park¹, and Byung-Seon Choi²

¹Sunkwang T&S

²Korea Atomic Energy Research Institute (KAERI)

*bhkim@sktns.co.kr

A large amount of concrete radioactive waste is generated during the decommissioning of nuclear facilities, including nuclear power plants, and it is expected that the radioactive waste management expenses will be huge. In order to reduce the concrete radioactive waste, a decontamination or removal process is required, but working on concrete may present a risk of worker exposure in a high-radioactive space. Therefore, in this study, a remote control integrated decontamination equipment that can reduce concrete radioactive waste and ensure the safety of workers during the concrete decontamination process in a high-radioactive space was developed.

The integrated decontamination equipment consists of remote movement, automatic surface contamination measurement, automatic surface decontamination and debris/dust removal systems.

The remote movement system generates ‘mapping data’ of topographic features for the work space and ‘location data’ that coordinates the location of the integrated decontamination equipment through LiDAR (Light Detection and Ranging) sensor and SLAM (Simultaneous Localization And Mapping) technique. The user can check the location of the integrated decontamination equipment through ‘location data’ outside the work space, and can move it by remote control through wired/wireless communication.

The automatic surface contamination measurement system uses a radiation detector and an automatic measurement algorithm to generate ‘surface measurement data’ based on the measurement distance interval and measurement time set by the user. ‘Surface measurement data’ is combined with ‘location data’ to create a visualized map of radioactive contamination, and users can intuitively realize the location and degree of contamination based on the map.

The automatic surface decontamination system uses a laser and an automatic removal algorithm to decontaminate the concrete surface. Concrete debris and dust generated during this process were collected by the debris/dust removal system, minimizing waste generation and radiation exposure due to secondary pollution.

The integrated decontamination equipment developed through this study was applied with technologies that reduced radioactive concrete waste and ensured the safety of workers. If technology verification and on-site applicability review are performed using concrete specimens simulating nuclear power plant or similar environments, it is reasoned to contribute to the domestic and overseas decommissioning industry.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 1510300110).

Keywords: Decommissioning, Concrete Decontamination, Radioactive Waste Reduction, Remote Control, Mapping

Upcycling of Radioactive Wastes to Foam Filter for Performance Improvement Using Sacrificial Material

Jang-Yul Kim*, Yong-Soo Jung, and Gangho Lee

Research Institute of Industrial Science & Technology (RIST)

*jyulkim@rist.re.kr

The dismantlement of the Kori Unit 1 and Wolsong Unit 1 nuclear power plants is scheduled. Since about 40% of the cost of dismantling nuclear power plants is the cost of disposing of generated wastes, it is important to secure recycling technologies. Among them, low and intermediate level radioactive wastes are made of porous filters and adsorbent materials of ceramic foam to remove nuclides such as C-14, I, and Xe generated during nuclear dismantling.

In order to remove a large amount of nuclides, physical properties such as a specific surface area and porosity of a ceramic foam filter are important, however when a heat treatment temperature is increased to increase the strength of the filter, the nuclides removal ability is reduced. In order to remove a large amount of nuclides, physical properties such as a specific surface area and porosity of a ceramic foam filter are important, however when a heat treatment temperature is increased to increase the strength of the filter, the nuclides removal ability is reduced.

Therefore, in this study, the foam filter performance was improved by applying a sacrificial material to increase the specific surface area and porosity of the ceramic foam filter. The sacrificial material is burned out with polyurethane (PU) of the green filter before the heat treatment temperature to increase the strength of the ceramic foam filter so that it can be maintained as pores, thereby improving the specific surface area and porosity.

The sacrificial materials and melting temperature (T_m) reviewed in this study were anthracite (530~660°C), PMMA (160°C), Cellulose acetate (260~270°C), and aluminum particle (660°C), and their effect on the manufacture of foam filters was studied by applying this. The specific surface part and porosity of the foam filter were improved when anthracite and aluminum particle were added, and PMMA and Cellulose acetate, which are relatively low temperature melting points, were burned out at a temperature lower than PU, and thus their physical properties were not greatly affected. The physical properties and specific surface part and porosity of ceramic foam filters manufactured using various sacrificial materials will be discussed.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 20201520300070).

Keywords: Low and Intermediate-level, Radioactive Waste, Concrete Waste, Foam Filter, Upcycling

Concept Designs of a Sampler Device for Taking Representative Samples From Waste Packages of High Integrity Containers (HICs)

Hang-Rae Cho^{1,*}, Sang-June Park¹, Young-Hwan Hwang², and Su-Il Bang¹

¹Korea Hydro & Nuclear Power (KHNP)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hrcho6795@khnp.co.kr

Radioactive spent resin and concentrate waste powder generated from the primary system of nuclear power plants (NPPs) should be treated and disposed of in the form of solidified products or high integrity container (HIC) packages. We are preparing for the application of polymer concrete high integrity containers (PC-HICs) that has been approved for disposal and field application after going through the disposal suitability review of the repository operator and the license review process of the regulatory body. A reliable assessment of nuclide inventory in waste drum is required for the disposal of the radioactive waste drums, and the representative samples should be collected for both the indirect (non-destructive assessment based on the scaling factor, average radioactivity concentration, etc.) and direct (destructive analysis) evaluation of the difficult-to-measure (DTM) nuclides. It is important to secure the representativeness of samples for reliable and accurate evaluation of radionuclide inventory and approval of methodologies for highly radioactive waste such as spent resin and concentrate waste powder, and in order to secure the radiation safety of the sampling workers and representativeness of the samples, a remote sampling method is required with excellent convenience and safety and sufficient representativeness of the sample. The simple sampling device used in the past to collect samples for the scaling factor does not have a remote control function, so high-radiation samples must be collected within a very short time and it is difficult to obtain sufficiently representative samples due to structural characteristics that cannot collect the entire sample in the axial direction of the package. Therefore we developed concept designs for a remote sampling device that can satisfy both sample representativeness, operator convenience and safety.

Keywords: Remote Sampling Device, Concept Design, PC-HIC, Representative Sample

The Refinement System Development for Lubricant Oil Waste in the Form of Organic Compounds Contaminated With Radioactive Materials

Hui-Gyeong Kim*, Yong-Ho Hong, Dong-Sik Jin, Sang-Soo Kwak, and Dong-Cheol Kim

Nuclear Environment Technology Co., Ltd.

*kimhphp@netec.co.kr

Lubricant oil waste contaminated with radioactive materials generated at nuclear facilities can be disposed of as industrial waste in accordance with self-disposal standards if only radioactive materials are removed. Lubricant oil used in nuclear facilities consists of oil of 75-85% and additives of 15-25%, and lubricant oil waste contains heavy metals, carbon, glycol, etc. In addition, lubricant oil waste from nuclear facilities contains metallic gamma-ray emission radionuclides including Co-60, Cs-137 and volatile beta-ray emission radionuclides such as C-14 and H-3, which are not present in lubricant oil waste from general industries and these radionuclides must be eliminated according to the Atomic Energy Act.

In general industries, the wet treatment technologies such as acid-white soil treatment, ion purification, thin film distillation, high temperature pyrolysis, etc. are used as the refining technology of lubricant oil waste, but it is difficult to apply these technologies to nuclear industrial sites due to restrictions related with controlling the generation of secondary radioactive waste in sludge condition containing radionuclides of metal components, and limiting the concentration of volatile radioactive elements contained in refined oil to be below the legal threshold.

In view of these characteristics, the refinement system capable of efficiently refining and treating lubricant oil waste contaminated with radioactive materials generated in nuclear facilities has been developed. The treatment process of this R&D system is as follows. First, the moisture in the radioactive lubricant oil waste pretreated through the preprocessing system is removed by the heated evaporating system, and the beta-emission radionuclides of H-3 and C-14 can be easily removed in this process. Second, the heated lubricant oil waste by the heated evaporating system is cooled through the heat exchanging system. Third, the particulate matters with gamma-ray emission radionuclides are removed through the electrostatic ionizing system. Forth, the lubricant oil waste is stored in the storage tank and the purified lubricant oil waste is discharged to the outside after sampling and checking from the upper, middle and lower positions of the lubricant oil waste stored in the storage tank.

Using this R&D system, it is expected that the amount of radioactive waste can be reduced by efficiently refining and treating lubricant oil waste in the form of organic compounds contaminated with radioactive materials generated in nuclear facilities.

Keywords: Lubricant Oil Waste, The Refinement System, Radioactive Material, Organic Compound

Automatic Waste Position Location and Its Information Administration for Safe Disposal Site Management Using Beacon

Sungjong Kim*, Kweonwoo Sohn, and Changhwan Cho

mirae-en

*sungjong.kim@mirae-en.co.kr

To efficiently manage the waste packages like drums, it is meaningful to utilize the data of placement and emplacement of disposed waste in geological storage. For the transparent and real-time management of disposal data, both technical as well as administrative factors must be included. To this end, MIRAE-EN Co., Ltd. has developed a radioactive waste tracking and management system (m-trek[®] v1.0) through radioactive waste management life cycle which is supported by KETEP.

Enhancing the functional features of m-trek[®], IoT-based drum location measuring and data of those drums, such as position, radionuclides, activity, and dose etc., are to be collected and monitored through data modeling and visualization, which might be utilized in emplacing the loaded drums according to specifically certain criteria of internal and external data of disposal site.

Position measuring using Beacon utilizes Received Signal Strength Indicator (RSSI) to locate the correct position in 3D area. Since RSSI is affected by the surrounding environment, it is required to corrective optimization. In addition, error and deviation of previously applied Gaussian filter method, was corrected and improved through AI learning model. According to this location information and those data, the prototype in future provides the visualization of drum position and its relevant data for administrative purpose such as monitoring, emplacement and other management policy.

Keywords: Waste Tracking, Beacon, IoT, Position Measurement, Visualization

A Safety Assessment of a Radioactive Waste Disposal Facility: A Well Scenario

Mi-Seon Jeong*, Dongkook Park, Soo-Gin Kim, and Jae-Yeol Cheong

Korea Radioactive Waste Agency (KORAD)

*jeongms@korad.or.kr

A radioactive waste disposal facility needs to be developed in a way to protect present and future generations and its environment. A safety assessment is implemented for normal and abnormal scenarios and human intrusion scenarios as a part of a safety case in developing a disposal facility for the radioactive waste. The human intrusion scenarios include a well scenario which takes into account various potential exposure groups (PEGs) who use a groundwater well contaminated with radionuclides released from the disposal facility.

It is observed that a pumping rate has a negative correlation with the biosphere dose conversion factor (BDCF) in the well scenario. C-14 is shown to be a key radionuclide in the well scenario, and a special model based on the carbon cycle is applied for C-14. For Tc-99, an adsorption coefficient should be adjusted to be suitable for the site.

The safety assessment for the radioactive waste disposal facility is successfully carried out for the well scenario. However, it is observed that site-specific models need to be developed and site-specific input data need to be collected in order to avoid unnecessary conservatism.

Acknowledgements

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (Ministry of Trade, Industry and Energy (MOTIE)) (No. 2021040101003C).

Keywords: Disposal Facility, Human Intrusion, Well Scenario, Input Data, Model

Preliminary Modeling to Characterize the Impact of Surrounding Structures on Neutron Activation in Unshielded Cyclotron Facility Using MCNP With FISPACT

Hyun Seok Song and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Important medical radionuclides for Positron Emission Tomography (PET) are producing using cyclotrons. There are about 1,200 PET cyclotrons operated in 95 countries based upon IAEA database (2020). Besides, including PET cyclotrons, demands for particle accelerators are continuously increasing. In Korea, about 40 PET cyclotrons are in operating phases (2020). Considering design lifetime (about 30-40 years) and actual operating duration (about 20-30 years) of cyclotrons, there will be demands for decommissioning cyclotron facilities in the near future.

PET cyclotron produces radionuclides by irradiating accelerated charged particles to the targets. During this phase, nuclear reactions ($^{18}\text{O}(p,n)^{18}\text{F}$ etc.) produce secondary neutrons which induce neutron activation of accelerator itself as well as surrounding infrastructures (the ancillary subsystems, peripheral equipment, concrete walls etc.). Generally, experienced cyclotron personnel prefer an unshielded cyclotron because of the repair and maintenance time. In unshielded cyclotron, water cooling systems, air compressor, and other equipment and structures could be existed for operating purposes. Almost all the equipment and structures are consisted of steel, and these affect neutron distribution in vault especially thermal neutron on the concrete wall. In addition, most of them can be classified as very low level radioactive wastes by Nuclear Safety and Security notice (NSSC Notice No. 2020-6). However, few studies were estimating radioactivity concentrations (Bq/g) of surrounding structures using mathematical calculation/simulation codes, and they were not evaluating the effect of surrounding structures on neutron distribution.

In this study, by using computational neutron transport code (MCNP 6.2), and source term calculation code (FISPACT- II), we evaluated effect of the interaction between surrounding structures (including surrounding equipment) and secondary neutrons. Discrepancies of activation distribution on/in concrete wall will be occur depending on thickness of structure, distance between structures and walls, and consideration of interaction between structures and neutrons.

Throughout this study, we could find that the influence of those structures can affect neutron distribution in concrete walls even if, thickness of the structure was small. For estimating activation distribution in unshielded cyclotron vault more precisely, not only considering cyclotron components and geometry of target, but also, considering surrounding structures will be much more helpful.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of the Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2021040101002B).

Keywords: Unshielded Cyclotron, Decommissioning, Activation Distribution, Surrounding Structure, MCNP 6, FISPACT

Status Update for Contaminated Water Management in Fukushima Daiichi NPP

Sunghoon Hong*, Seok-Ju Hwang, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sunghoon.hong@khnp.co.kr

Recently, Japan's government has announced Tokyo Electric Power Company's plan to discharge contaminated water stored from the tanks of the Fukushima Daiichi nuclear power plant site into the sea. 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 1,311,736 m³ (as of August 18, 2022). The amount of contaminated water is estimated same as Tokyo dome volume.

Under the sea discharge plan, the contaminated water will be diluted in seawater more than 100 times, and tritium concentration lowered 1/7 of the drinking water standard set by the World Health Organization (10,000 Bq/liters). The diluted water will then move through an undersea tunnel and be discharged about 1 kilometer off the coast.

Keywords: Fukushima Daiichi NPP, Contaminated Water, ALPS, Sea Discharge

Review of New Regulatory Guidelines in Relation to the Off-site Dose Calculation

Deuk-Man Kim*, Seok-Ju Hwang, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*get10000laver@khnp.co.kr

In 2022, new regulatory guidelines were announced in relation to the off-site dose calculation (ODC), and accordingly, measures to improve the off-site dose calculation program (ODCP), k-dose60, were reviewed. The main consideration is, first, that if multiple nuclear facilities are operated on the same site, the boundaries of the restricted areas shall be set as the overlapping outer boundaries of the restricted areas determined by calculation for each nuclear facility. Second, the external exposure caused by direct radiation from a number of nuclear facilities in the same site must be partially or fully applied depending on the facility and site characteristics. Third, the dose conversion coefficient should be evaluated by checking whether the effect of the daughter nuclides is properly reflected. Fourth, the soil contamination period is a factor to consider that radioactive substances deposited on the surface, such as particulate nuclides, affect residents over a long period of time. Fifth, due to the recent construction of Shin-Kori Units 5 and 6, there is a change in the site boundary of the Kori/Saeul site, so as the site boundary is expanded, it is required to add an exposure dose assessment point due to gas effluents and change the exposure dose assessment point according to crop intake. Therefore, through this study, the direction for improving the ODOP will be prepared by reviewing the recent revision of the regulatory guidelines.

Keywords: Off-site Dose Calculation Program, K-DOSE60, Guideline

Preliminary Consideration of Radiological Risk Associated With Transport of Spent Fuel and Radioactive Waste in Environmental Impact Assessment for Korean NPPs

Younghoon Hwang and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

The U.S. Nuclear Regulatory Commission (NRC) states that every environmental report prepared for the licensing stage of a Pressurized Water Reactor shall contain a statement concerning risk during the transportation of fuel and radioactive wastes to and from the reactor. Thus, the licensee should ensure that the radiological effect in accidents, as well as normal conditions in transport, do not exceed certain criteria or be small if cannot be numerically quantified. These are specified in 10 CFR Part 51 and applied in NUREG-1555 Supplement 1 Revision 1, which deals with Environmental Standard Review Plan.

Corresponding regulations in Korea would be the Nuclear Safety and Security Commission Notice No. 2020-7. In Appendix 2 of the Notice, guides on the radiological environmental report for production and utilization facilities, spent nuclear fuel interim storage facilities, and radioactive waste disposal facilities. In this guide, unlike the regulations in the U.S., there are no obligations for radiological dose assessment for workers and public during the transportation.

Therefore, overall regulations and their legal basis related to risk assessment during transportation conducted for the environmental report in the U.S. were analyzed in this study. On top of that, through the comparison with regulations in Korea, differences between the two systems were figured out. Finally, this study aims to find the points in terms of assessing transport risk to be revised in the current regulatory system in Korea.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of the Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 20203210100240).

Keywords: Environmental Report, Transportation, Radiological Effect, Risk Assessment

An Overview of the Revision History of Assessment Methodology of Radioactive Source Term for Normal Operation of Light Water Reactors

Ji-Young Song* and Kwan-Hee Lee

Korea Institute of Nuclear Safety (KINS)

*jysong@kins.re.kr

In Korea, the NUREG-0017 methodology based on realistic model for reactor coolant concentrations are used to estimate the annual radioactive effluent releases for normal operation of nuclear power plant. The realistic model to estimate the radionuclide concentrations in reactor coolant is formulated as a standard, ANSI/ANS-18.1. This standard has provided a set of the reference radionuclide concentrations and adjustment factors for estimating the radioactivity in the principal fluid systems of target plant. Since ANSI/ANS-18.1 was first published in 1976, it was revised in 1984, 1999, 2016, and most recently in 2020. Therefore, this study analyzed revision history of assessment methodology of radioactive source term of light water reactors, which is ANSI/ANS-18.1.

Assessment methodology of radioactive source term given ANSI/ANS-18.1 is by using radionuclide concentrations for reactor coolant and steam generator fluid of the reference plant and adjustment factors, which is modifying radioactive source term according to differences in design parameters between reference plant and target plant. There are three type of reference plant: PWR with u-tube steam generator, PWR with once-through steam generator, and BWR. This study analyzed for PWR with u-tube steam generator. Although the standard was revised, evaluation methodology and formula of adjustment factor have been retained, but some of items have been revised. First revision item is reduction of the number of radionuclides and decrease of radioactive concentration in reactor coolant. In the 1976 version of the standard, there were 71 target radionuclides, but the target nuclides have reduced to 57 in 1984 and 56 after 1999. In the case of radioactive concentration in reactor coolant, as the version of standard was updated, the radioactive concentration of 18 nuclides in 1984, 14 nuclides in 1999, and 25 radionuclides in 2016 was decreased. Most of the radionuclides with decrease radioactivity concentration were fission product, it is resulted from improvement of nuclear fuel performance. Second revision item is change of adjustment factors. After the revision in 2016, the adjustment factors for zinc addition plants using natural or depleted zinc are changed.

This study analyzed revision history of evaluation methodology of radioactive source term of light water reactors. Furthermore, result of this study will be contributed to the improvement of understanding of assessment methodology and revision history for the radioactive source term.

Keywords: Radioactive Source Term, Adjustment Factor, System Design

Trends of ^3H Decontamination Technology and Design of Pilot Hybrid Technology Facilities for ^3H Contaminated Water Treatment

Jiyang So, Doocheol Kim, Seonggeon Kim, Sangwook Lee, Seongjoo Kang, Minho Kwak, and Seungil Kim*

Elim-Global

*seungil.kim@elim-global.com

In the field of ^3H decontamination technology, the number of patent applications worldwide has been steadily increasing since 2012 after the Fukushima nuclear accident. In particular, Japan has a relatively large number of intellectual property rights in the field of ^3H processing technology, and it seems to have entered a mature stage in which the growth rate of patent applications is slightly reduced. In Japan, tritium is being decontaminated through the Semi-Pilot-class complex process (ROSATOM, Russia) using vacuum distillation and hydrogen isotope exchange reaction, and the Combined Electrolysis Catalytic Exchange (CECE, Kurion, U.S.) process. However, it is not enough to handle the increasing number of HTOs every year, so the decision to release them to the sea has been made. Another commercial technology in foreign countries is the vapor phase catalyst exchange process (VPCE) in operation at the Darlington Nuclear Power Plant in Canada. This process is a case of applying tritium exchange technology using a catalyst in a high-temperature vapor state.

The only commercially available tritium removal technology in Korea is the Wolseong Nuclear Power Plant's Removal Facility (TRF). However, TRF is a process for removing HTO from D_2O of pure water, so it is suitable only for heavy water with high tritium concentration, and is not suitable for seawater caused by Fukushima nuclear power plant's serious accident, and surface water and groundwater contaminated by environmental outflow of tritium.

Until now, such as low-temperature decompression distillation method, water-hydrogen isotope exchange method, gas hydrate method, acid and alkali treatment method, adsorption method using inorganic adsorbent (zeolite, activated carbon), separator method using electrolysis, ion exchange adsorption method using ion exchange resin, etc. have been studied as leading technologies for tritium decontamination. However, any single technology alone has problems such as energy efficiency and processing capacity in processing tritium, and needs to be supplemented.

Therefore, in this study, four core technologies with potential for development were selected to select the elemental technology field of pilot facilities for treating tritium, and specialized research teams from four universities are conducting technology development. It was verified that, although each process has different operating conditions, tritium removal performance is up to 60% in the multi-stage zeolite membrane process, 30% in the metal oxide & electrochemical treatment process, 43% in the process using hydrophilic inorganic adsorbent, and 8% in the process using functional ion exchange resin. After that, in order to fuse with the pretreatment process technology for treating various water quality tritium contaminated water conducted in previous studies, the hybrid composite process was designed by reflecting the characteristics of each technology. The first goal is to create a Pilot hybrid tritium removal facility with 70% tritium removal efficiency and a flow rate of 10 L/hr, and eventually develop a 100 L/hr flow tritium removal system with 80% tritium removal efficiency through performance improvement and scale-up. It is also considering technology for the post-processing process in the future.

Acknowledgements

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

Keywords: Tritium Decontamination, Pilot Facility, Adsorption of Tritium, Ion Exchange Resin, Inorganic Adsorbent, Electrolysis

Review of Licensing Process of the 2nd Phase Low Level Radioactive Waste Disposal Facility in Gyeongju

Yohan Kim, Eunjin Seo*, Jinyong Park, Chan Woo Jeong, Sangmyeon Ahn, Jeheon Bang, Jungjin Kim,

Sang-Ho Lee, Jinmo Ahn, and Sangsu Park

Korea Institute of Nuclear Safety (KINS)

*k697sej@kins.re.kr

This study introduces the licensing process carried out by the regulatory body for construction and operation of the 2nd phase low level radioactive waste disposal facility in Gyeongju. Also, this study presents the experience and lessons learned from this regulatory review for preparing the license review for the next 3rd phase landfill disposal facility.

Korea Radioactive Waste Agency (KORAD) submitted a license application to Nuclear Safety and Security commission (NSSC) on December 24, 2015 to obtain permit for construction and operation of the national engineered shallow land disposal facility at Wolsong, Gyeongju. NSSC and Korea Institute of Nuclear Safety (KINS) started the regulatory review process with an initial docket review of the KORAD application including Safety Analysis Report, Radiological Environmental Report and Safety Administration Rules. After reflecting the results of the docket review, the safety review of revised 10 application documents began on November 29, 2016. Total 856 queries and requests for additional information were elicited by thorough technical review until November 16, 2021. As the Gyeongju and Pohang earthquakes occurred in September 2016 and November 2017, respectively, the seismic design of the disposal facility for vault and underground gallery was enhanced from 0.2 g to 0.3 g and the site safety evaluation including groundwater characteristics was re-investigated due to earthquake-induced fault. Also, post-closure safety assessments related to normal/abnormal/human intrusion scenarios were re-performed for reflecting the results of site and design characteristics. Finally, NSSC decided to grant a license of the 2nd phase low level radioactive waste disposal facility under the Nuclear Safety Laws in July 2022.

This study introduces important issues and major improvements in terms of safety during the review process and presents the lessons learned from the experience of regulatory review process.

Keywords: Licensing Process, Engineered Shallow Land Disposal Facility, Low Level Radioactive Waste

Study on Test Requirements Applied to Immobilized Solid Form

Jisoo Yoon* and Ji-Hoon Lee

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*jisooYoon1@khnp.co.kr

The disposal criteria of the domestic LILW disposal facility specifies that fluidized substances such as the spent resin, the evaporator bottom should be solidified in a physically stable solid form, such as cementation and polymerization. And the solidified form applies requirements for compressive strength, immersion test, thermal circulation test, radiation irradiation test, leaching test, and free standing water measurement test. On the other hand, it is specified that immobilization is applied to wastes with a total radioactivity concentration of more than 74,000 of radionuclides with a half-life exceeding 20 years among non-homogeneous wastes such as spent filters and DAW, but the test requirements are not applied. Nevertheless, it is necessary for waste generator to establish quality control standards for the manufacture of immobilized solid form through reviewing overseas cases and domestic regulations and technical standards.

The test requirements for solidified solid form require measurement of structural stability (compressive strength, immersion, thermal cycling, irradiation test), leachability (leaching test), and free standing water measurement. A characteristic of the immobilized solid form is that it is not mixed with the waste and that the cement medium surrounds the waste. Therefore, the structural soundness is higher than that of the solidified solid mixed with waste. In addition, even when in contact with water, the cement medium blocks the contact between waste and water, thereby preventing the spread of radionuclides.

Therefore, considering the characteristics of these immobilized solid form, compressive strength test and free standing water measurement are applied for structural soundness. For other tests, it is determined that application is unnecessary.

Keywords: Immobilization, Solidification, Inhomogeneous Waste, Solid Form, Spent Filter

Benefit Analysis for Melting of Metallic Intermediate Level Waste

Moonoh Kim^{1,*}, Yoonji Choi², and Jeonghyun Yoon²

¹KONES Corp.

²Korea Radioactive Waste Agency (KORAD)

*mokim@kones21.com

The treatment of radioactive waste by melting has been mainly discussed with low-level waste (LLW). Considering that a large amount of waste in RV or RVI is intermediate-level waste (ILW), however, it is necessary to examine the possibility of treatment by melting of ILW. Different from LLW, melting of ILW with a high content of long-lived nuclides would lead to no free release, but has advantages in volume reduction, homogenization, and delay of release. In this paper, the possibility of melting as an alternative technology for the treatment of ILW in the future is reviewed by analyzing the benefits generated by melting ILW in the following aspects: 1) Similar to melting techniques of LLW, those of ILW are mostly based on well-known techniques, but it is necessary to review the feasibility of performing operations such as removal of solidified melt using remote equipment in abnormal situations such as loss of electricity. 2) It is necessary to specify radiation limits for the melting operation unless the ILW melting operation technique can guarantee that the risk of abnormal occurrence is very low. The main quantified radiation parameter is the ingot dose rate, which of 10 mSv/h is considered more reasonable. 3) Although the treatment of ILW by melting leads to a reduction in volume, the main characteristics of the waste still remain, and no waste can be disposed of for free release. Thus, the main potential benefits are improved long-term safety and reduced waste volume. 4) Reducing the surface-to-volume ratio of the molten material could reduce the amount of corrosive material per unit time and, consequently, increase long-term safety. Its effect on long-term safety is difficult to quantify precisely as it depends on several factors, such as the geometry of the original component or whether radionuclides were distributed on the surface of the original component or the induced radioactivity. 5) The volume reduction of ILW is estimated to be reduced by about 1/4 compared to the generated amount when assuming a disposal volume reduction factor of 3 and considering the dose reduction due to radioactive decay after long-term storage, however, due to the lack of knowledge about non-hazardous facility alternatives, it is difficult to evaluate cost-benefit. This is heavily influenced by both the final volume reduction and the potential to reduce the complexity of the repository's technical barriers.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No.20211510100040).

Keywords: ILW, LLW, Melting, Metal, Benefit

A Study on Prediction of Annual Disposal Capacity for Gyeongju Radioactive Waste Repository Using System Dynamics Modeling

Yuwhwa Son*, Jachul Ha, Chunhyung Cho, and Hyunchul Kwon

Korea Radioactive Waste Agency (KORAD)

*slowstrider@korad.or.kr

Gyeongju radioactive waste repository has been operated to dispose low and intermediate level radioactive waste in Korea since 2016. Currently, only deep geological disposal facility (1st) is in operation, surface disposal facility (2nd) is scheduled to operate from 2024. As a result, the annual amount of radioactive waste that can be disposed of at deep geological disposal facilities and surface disposal facilities is almost determined. According to this result, it was possible to derive the total annual disposal amount to dispose of all radioactive waste at the Gyeongju repository after landfill disposal facility (3rd) construction. To evaluate it, a predictive model has been designed and radioactive waste generation, storage, and disposal data were input. The predictive model is based on system dynamics, which is useful to analyze the correlation between input variables. As a result of analysis, radioactive waste generation amount and maximum annual radioactive waste disposal were predicted to reach 741,615 drum and 17,030 drum per year respectively. From these results, it seems that the expansion of radioactive waste acceptance system or temporary storage is necessary.

Acknowledgements

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

Keywords: Radioactive Waste, Annual Disposal Capacity, Predictive Model, System Dynamics

A Case Study for Clearance of Volumetric Contaminated Wood Wastes

Yunjeong Hong* and WonHyuk Jang

Korea Atomic Energy Research Institute (KAERI)

*hongyj@kaeri.re.kr

As the management procedure for self-disposal wastes stored in the radiation controlled area within the Korea Atomic Energy Research Institute (KAERI) have been established, and the types and quantities of wastes are increasing. In order to carry out the disposal of wastes with various generation histories, we expanded the processing range from surface contaminated waste, which was already in progress, to volumetric contaminated waste. In this paper, a case study of self-disposal of volumetric contaminated radioactive waste for which final disposal has been completed is described.

In order to carry out of self-disposal of volumetric contaminated waste, it is important to collect representative samples and prove their representativeness. Based on this, a treatment plan was established after reviewing the history of the waste to be disposed of, and the treatment work was carried out as follows; waste collection, classification by size and shape, radiation (activity) measurement, sampling of representative samples, radioactivity concentration analysis, notification to regulatory bodies and question-and-answer, final disposal. The waste is judged have no potential for contamination because the polywood used to set the flat floor between the steel frame and floorboards in the experimental greenhouse didn't come into contact with radioactive material. However, due to the conservative approach to the presence or absence of contamination, the treatment plan was established assuming volumetric contaminated waste. The type of waste is single wood, and the major contaminating radionuclides are Sr-85 and Cs-137. After the waste was collected and sorted by size and shape, it was weighed and a representative sampling amount and sampling method were set up. A direct method of surface contamination was performed on the entire area, and the representative sample was divided into three groups of homogenized population samples using the trisection method, with three points (upper/middle/below) were sampled at a 200:1 ratio, and radioactivity concentration analysis was conducted. After confirming that the concentration was below the allowable concentration for self-disposal, the final disposal was completed after receiving approval after reporting to the regulatory body.

As a result of radioactivity concentration analysis of representative samples, the maximum radioactivity concentration for each nuclide was Sr-85: < MDC (0.00178), Cs-137 : 0.00183 Bq/g (Sr-85 : 1 Bq/g, Cs-137 : 0.1 Bq/g), which meets the nuclide allowable concentration standard. It was confirmed that the total maximum fraction of 0.02 Bq/g satisfies the criteria (In the case of mixed nuclides, the sum of the fraction is less than 1).

This paper introduces the establishment and implementation of self-disposal procedures based on the experience of self-disposal of radioactive waste with volumetric contaminants, and is going to utilize it as a basic material for self-disposal of radioactive waste with volumetric contaminants that will continue in the future and contribute to the reduction of radioactive wastes.

Keywords: Clearance, Radioactive Waste, Radioactivity Concentration, Surface Contamination, Volumetric Contamination

Examination for Repackaging Method of Intermediate Level Waste

Moonoh Kim^{1,*}, Yoonji Choi², and Jeonghyun Yoon²

¹KONES Corp.

²Korea Radioactive Waste Agency (KORAD)

*mokim@kones21.com

Treatment methods such as interim storage and immobilization are being considered to dispose of intermediate level waste (ILW), but some wastes that have been treated in the past may require re-packaging. Re-packaging means to cover repackaging of waste that has already been packaged in a waste container and re-packaging is required for the following reasons: loss of shielding or containment, damage to external handling features, package out-of-specification, insufficient records and external policy. The re-packaging includes various methods such as non-intrusive treatment, overpacking of waste package, external treatment of waste container, repair waste container, injection of stabiliser, disassemble waste package, high temperature process, and dissolve waste package. The purpose of this paper is to evaluate the re-packaging possibility for various wastes by identifying the main re-packaging methods among the above various re-packaging methods. 1) Disposal outside of the waste container is a viable technique for most packages, as coating with a portable spray gun for low dose rate packages or remotely using a robotic arm for high dose rate packages. 2) Waste container repair is divided into welding repair and patching of waste container according to the degree of damage. Weld repair and patching are important techniques that can be used to add additional shielding, repair damaged areas, and improve the integrity of lifting gears that may not be compliant. 3) In general, disassembly of waste packages has been applied to loose drummed waste. Packages and waste forms are physically disassembled, reduced in size, and placed in different new packages. For practical solution, grouted waste is repackaged by cutting using proprietary equipment such as diamond saws, wire saws, core drilling and rupture techniques. 4) High-temperature process involves cutting the waste package and placing the pieces in a hot bath of inorganic liquid or molten metal, and the process is applicable to all waste types. However, treatment of all gases produced, compliance with waste types and acceptance criteria. Finally, dissolving waste packages, which is generally considered impractical due to the variety of chemicals and radionuclides present in ILW, is a process that is easier to perform on raw ILW than conditioned waste. An example of waste being re-packaged is when old drummed waste is recovered from an old storage facility and the waste needs to be re-packaged into a form that meets modern standards for interim storage and disposal.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. 20211510100040).

Keywords: ILW, LLW, Repackaging, Package, Container

Corrected Off-Gas Illustrative Model for 6 MW Molten Salt Reactor

Ju Ho Lee*, Seok-Min Hong, and ChangHwa Lee

Korea Atomic Energy Research Institute (KAERI)

*jhlee835@kaeri.re.kr

Molten Salt Reactor, which employs molten salt mixture as fuel, has many advantages in reactor size and operation compared to conventional nuclear reactor. In developing Molten Salt Reactor, Off-gas system should be properly designed since the fission products in off-gas accelerates the corrosion in reactor structure materials and deteriorates the purity of liquid fuel. The design of off-gas system therefore requires the preliminary study of the behavior of evolved fission products in off-gas units and the development of off-gas model is crucial in developing such system. In this study, we corrected the off-gas illustrative model proposed by ORNL (Nuclear Engineering and Design, vol 385(15) 111529, 2021) by employing physically consistent concept of capture rate of fission product and hold-up. For the application of the corrected off-gas model to Chloride-based 6 MW Molten Salt Reactor, major fission products were firstly determined from OpenMC based neutronics calculation and chain reaction related to the major fission products were defined. Based on these data, the holdup behavior of fission products in off-gas units (decay tank, caustic scrubber, Halide trap, H₂O trap and charcoal bed) were investigated.

Acknowledgements

This work was supported by the KAERI Institutional Program (Project No. 522330-22).

Keywords: MSR, Off-gas, Hold-up, Decay, Residence Time

The Characteristics of Solidified Sludge According to the Distribution of Ferro Frit Powder in Sludge

Ki Joon Kang, Sia Hwang, and Hee Reyoung Kim*

Ulsan National Institute of Science and Technology (UNIST)

*kimhr@unist.ac.kr

In nuclear power plant, there were many contaminated tanks dispose of radioactive fluid waste. These tanks are made of stainless-steel, and corrosion can occur when tanks are exposed to radioactive fluid waste containing moisture for a long time. Therefore, those sludge waste including radionuclide should be collected, solidified, and disposed of. If sludge can be melted, sludge can be easily solidified. However, melting points of sludge components (Fe_2O_3 , NiO , Cr_2O_3) are very high as 1565, 1955, and 2435, respectively. Therefore, melting sludge is difficult. If a solidification auxiliary material such as cement or asphalt is used to help solidify, solidification can easily occur, but cement and asphalt are vulnerable to heat. Vitrification using glass material can be solidification method, but the waste loading ratio of glass material is higher than 50%. High waste loading ratio is weakness in terms of volume reduction of waste. In this study, ferro frit powder (Na_2O , K_2O , CaO , Al_2O_3 , B_2O_3 , SiO_2 , ZnO) is used as solidification auxiliary material. When ferro frit powder mixed with sludge material are melted in sludge material, melted ferro frit powder can stick sludge material and can solidify sludge material without melting. Sludge can be solidified by using ferro frit powder with a smaller waste loading ratio than the vitrification method. However, since the waste loading ratio of the solidification auxiliary material is small, if ferro frit powder is not uniformly distributed between sludge powder, solidification may not be performed properly. Although the mixing ratio between sludge and ferro frit in solidified sludge is same, when the distribution of ferro frit powder in sludge is non-homogeneous, the difference in chemical and physical characteristics as compressive strength and leaching resistance can be observed in solidified sludge. As the ferro frit mixing ratio in the site where ferro frit exists was relatively high, the melting point of the mixed powder (sludge+ferro frit) decreased, and the mixed powder could not maintain its shape and melted. In the case of the area where ferro frit does not exist, since only the stainless-steel oxide sludge exists, sludge was not melted, and the shape was maintained. However, it was confirmed that the leaching resistance was lowered by visually observing the color change of the leachate within a short period of time (about 2 hours) when solidified sludge was immersed in the leachate.

Acknowledgements

This study was supported by the Industrial Technology Innovation Program (20191510301210, Development of Dismantling Technology for the Volume Reduction of Radioactively Contaminated Tanks) funded by the Korea Institute of Energy Technology Evaluation and Planning (KETEP, Korea).

Keywords: Radioactive Sludge, Hot Isostatic Pressing, Capsulation, Solidification

5분과

제염해체(Oral)

Development of NPP Cost Estimation Program Through 3D and Unit Cost Factor (UCF)

Minhee Kim, Chang-Lak Kim*, and Sanghwa Shin

Kepeco International Nuclear Graduate School (KINGS)

*clkim@kings.ac.kr

3D modeling is a technology for representing real objects in a virtual 3D space or modeling and reproducing the physical environment. 2D drawings for viewing the existing building structure have limitations that make it difficult to understand the structure. By implementing this 3D modeling, specific visualization became possible. 3D technology is being applied in a wide range of pre-evaluation work required for nuclear decommissioning. In Slovakia, 3D modeling was applied to determine the optimal cutting strategy for the primary circuit before dismantling the VVER type Bohunice V1. In Japan, the Decommissioning Engineering Support System (DEXUS) program has been developed that incorporates VRDose, a decommissioning engineering support system based on 3D CAD models. Through this, the cutting length of the work object and the quantity of containers for packaging waste are calculated, exposure dose calculations in various dismantling scenarios, and cost estimation are performed. Korea also used 3D technology to evaluate the decommissioning waste volume for Kori Unit 1 and to evaluate the optimal scenario of the decommissioning process procedure for the research reactor Unit 1.

3D technology is currently being used in various pre-decommissioning evaluations for VVER, PWR, and research reactors. Overseas, a program that matches various decommissioning pre-evaluation tasks with cost estimation has also been developed. However, most 3D technologies are mainly used as a support system for dose evaluation and amount of decommissioning waste calculation. In this study, 3D modeling was performed on the PHWR structure, and physical and radiological information about the structure was provided. The information on the structure can present the unit cost for the work object by confirming the standard of the applied unit cost factor (UCF). The UCF presents the unit cost for repeated decommissioning operations. The decommissioning cost of the work object can be obtained by multiplying the UCF by the number of repetitions of the work. If the results of this study are combined with the process evaluation and waste quantity estimation performed in previous studies, it is judged that it will be helpful in developing an integrated NPP decommissioning program that requires preliminary evaluation of various tasks. In addition, it is judged that a clear cost estimation of the object to be evaluated will be possible by matching the 3D work object with the UCF.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130 and No. 20201510300130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: 3D Modeling, Unit Cost Factor, Decommissioning, Cost, Estimation

Development of a Platform That Enables Generation of Cutting Paths and Process Simulation for Decommissioning of Reactor Vessel Internals (RVI) Within a Virtual Environment

Jae-Seok Ki^{1,*}, Kyo-Chan Hwang¹, Ju-Ho Choi¹, Jeong Suh², and Ryoan-han Kim²

¹Sangmyung University

²Korea Institute of Machinery & Materials (KIMM)

*ceo@tritech.co.kr

The decommissioning process of Kori Nuclear Power Plant No.1, which was permanently suspended in 2017, various studies and attention on the decommissioning of nuclear power plants and waste management are being focused. In particular, decommissioning of high-risk facilities should take into account both safety and economic aspects. Small defects in the decommissioning process may lead to major disasters, and the resulting economic losses will cause enormous damage at the national level. In order to prevent such damage, various decommissioning process simulations within a virtual environment should be performed, and process errors and results should be collected and analyzed through simulation to derive the optimal decommissioning scenario as possible.

The platform introduced in this paper builds a virtual environment based on drawing and modeling data of Kori Nuclear Power Plant No.1 and automatically creates an optimized cutting path for dismantling the facility and internal structure, and simulates a cutting process similar to reality using Robot Arm. In addition, it is possible to derive and analyze a cutting process scenario by processing process results such as time required for work and cutting distance collected through simulation.

Acknowledgements

This paper was conducted with the support of the ICT-based Nuclear Safety Innovation Technology Development Project of the Korea Research Foundation in 2022.

Keywords: Virtual Environment, Simulation Program, Decommissioning Process, Digital Twin, Reactor Vessel Internals (RVI)

Development of 3D Imaging Equipment Available in Radioactive and Underwater Environments

Dongjun Hyun*, Sungmoon Joo, Ikjune Kim, Jaehyun Ha, and Jonghwan Lee

Korea Atomic Energy Research Institute (KAERI)

*dongjunn@kaeri.re.kr

3D imaging equipment is essential for automated robotic operations that cut radiologically contaminated structure and transfer segmented pieces in nuclear facility dismantling site. Automated dismantling operations using programmed robotic arms can make conventional nuclear facility dismantling operations much more efficient and safer, so dismantling technologies using robotic arms are being actively researched. Resolving the position uncertainty of the target structure is very important in automated robot work, and in general industries, the problem of position uncertainty is solved through the method of teaching the robot in the field, but at the nuclear facility dismantling site, the teaching method by workers is impossible due to activated target structures. Therefore, 3D imaging equipment is a key technology for a remote dismantling system using automated robotic arms at nuclear facility dismantling site where teaching methods are impossible.

3D imaging equipment available in radioactive and underwater environments is required to be developed for a remote dismantling system using robotic arms because most commercial 3D scanners are available in air and certain 3D scanners available in radioactive and underwater environments cannot satisfy requirements of the remote dismantling system such as measurement range and radiation resistance performance. The 3D imaging equipment in this study is developed based on an industrial 3D scanner available in air for efficient development. To protect the industrial 3D scanner against water and radiation, a housing is designed by using mirrors, windows and shieldings. To correct measurement errors caused by refraction, refraction model for the developed 3D imaging equipment is defined and parameter studies for uncertain variables are performed.

The 3D imaging equipment based on the industrial 3D scanner has been successfully developed to satisfy the requirements of the remote dismantling system. The 3D imaging equipment can survive up to a cumulative dose of 1 kGy and can measure a 3D point cloud in the air and in water with an error of less than 1 mm. To achieve the requirements, a proper industrial 3D scanner is selected, a housing and shielding for water and radiation protection is designed, refraction correction are performed. The developed 3D imaging equipment is expected to contribute to the wider application of automated robotic operations in radioactive or underwater environments.

Acknowledgements

This work was supported by a National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (RS-2022-00155255).

Keywords: 3D Imaging Equipment, Remote Dismantling System, Radiation Protection, Refraction Correction, Underwater

Development of Integrated Decontamination Process Using Supercritical Carbon Dioxide and Ethanol for Soil Contaminated With Cs, Sr, U

Seungil Ha, Kwangheon Park*, and Jinhyun Sung

Kyung Hee University

*kpark@khu.ac.kr

Korea faces decommissioning the nation's first commercial nuclear power plant, the Kori-1 and Wolsong-1 reactors. In addition, other nuclear power plants that will continue to operate will also face decommissioning over time, so it is essential to develop independent nuclear facility decommissioning and site remediation technologies. Among these various technologies, soil decontamination is an essential not only in the site remediation after the decommissioning of the highly radioactive nuclear facility, but also in the case of site contamination caused by an accident during operation of the nuclear facility. But the soil, which is a porous material, is difficult to decontaminate because radionuclides are adsorbed into the pores. Therefore, with the current decontamination technology, it is difficult to achieve the two goals of high decontamination efficiency and secondary waste reduction at the same time.

In this study, a soil decontamination process with supercritical carbon dioxide as the main solvent was presented, which has better permeability than other solvents and is easy to maintain critical conditions and change physical properties. Through prior research, a polar chelating ligand was introduced as an additive for smooth extraction reaction between radionuclides present as ions in soil and nonpolar supercritical carbon dioxide. In addition, for the purpose of continuity of the process, a candidate group of auxiliary solvents capable of liquefying the ligand was selected.

In this research evaluated the decontamination efficiency by adding the selected auxiliary solvent candidates to the supercritical carbon dioxide decontamination process, and ethanol with the best characteristics was selected as the final auxiliary solvent. In addition, based on the decontamination effect under a single condition of the auxiliary solvent found in the Blank Test process, the possibility of a pre-treatment leaching process using alcohol was tested in addition to the decontamination process using supercritical carbon dioxide. Finally, in addition to the existing Cs and Sr, the possibility of decontamination process was tested by adding U nuclides as a source of contamination.

As a result of this research, it is expected that by minimizing secondary waste after the process, waste treatment cost could be reduced and the environmental aspect could be contributed, and a virtuous cycle structure could be established through reuse of the separated carbon dioxide solvent. In addition, adding its own extraction capacity of ethanol used for liquefaction of solid-phase ligands is expected to maximize decontamination efficiency in the process of increasing the size of the process in the future.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT) (No. 2018M2B2B106563521).

Keywords: Supercritical Carbon Dioxide, Soil Decontamination, Site Remediation, Ethanol, Chelating Ligand, Ultrasonic Effect

Analysis of the Necessity of Building New Vessel for the Transport of Decommissioning Radioactive Waste

Sanghyeon Eom*, Moonoh Kim, and Woo Yong Kim

KONES Corp.

*sheom@kones21.com

The design life of the radioactive waste carrier, the CHEONG JEONG NURI, is in the year 2034, when the decommissioning of Kori Unit 1 is expected. As only IP-2 type transport containers (7.5-tons, 1.6 m (W) × 3.4 m (L) × 1.2 m (H)) can be loaded onto the CHEONG-JEONG-NURI, the radioactive decommissioning waste (RDW) transport containers neither of 35-tons maximum weight nor ISO type can be accommodated. Accordingly, either a new vessel (NV) to replace the CHEONG-JEONG-NURI or a change in the loading dock design of the CHEONG-JEONG-NURI is required.

In this study, the necessity of building a NV capable of accommodating the issued containers above is analyzed focusing, (1) the estimated building and operating costs of the NV, and (2) the economic feasibility of the NV 's RDW transportation scenarios.

Among bulk carriers, the CHEONG-JEONG-NURI was designed as handy-size ship type. It is operated reflecting various design requirements to satisfy the domestic/international legal requirements. To estimate the cost of the NV, the same vessel type and design criteria of the CHEONG-JEONG-NURI were considered.

The shipping price information of the Korea Ocean Business Corporation, as of August 2022, the building cost of bulk carrier Handysize (building NV type) is about USD 30 million. Considering domestic/overseas variables, such as future labor costs, international inflation, interest rate hike, etc., the building costs are expected to continuously rise.

Furthermore, vessel operation costs of crew labor, vessel, fuel, and insurance are incurred separately. Due to the increase in oil price, and wages of special positions, such as general seafarers and radiation safety managers, the NV's operating cost is expected to be about KRW 3.8 billion every year, which is about KRW 1.1 billion higher than that of the CHEONG-JEONG-NURI.

The expected total cost of building and operating the NV is about KRW 65 billion. Assuming the repayment period of the NV building cost is the same as that of the CHEONG-JEONG-NURI building cost reimbursement agency and analyzing the economic feasibility of the transport scenario of the NV built by adding up about KRW 3.8 billion of the operating cost, cost about KRW 880 million per voyage of the NV built is expected, which being KRW 620 million more than the current cost (KRW 260 million) per trip of the CHEONG-JEONG-NURI.

Therefore, transporting the RDW to the disposal facility through sustainable use of the CHEONG-JEONG-NURI (considering design life extension and design change) is evaluated as more appropriate than building NV.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No.20211510100040).

Keywords: CHEONG JEONG NURI, New Vessel, Marine Transport

A Study on Solidification of Uranium Contaminated Soil Waste Based on Glass-ceramization

Jaewoong Hwang^{1,2}, Jaseung Koo¹, and Keunyoung Lee^{2,*}

¹Chungnam National University (CNU)

²Korea Atomic Energy Research Institute (KAERI)

*lky@kaeri.re.kr

The decommissioning of nuclear-related facilities at the end of their design life generates various types of radioactive waste. Therefore, the research on appropriate disposal methods according to the form of radioactive waste is needed. This study is about the solidification of uranium contaminated soils that may occur on the site of nuclear facilities. A large amount of radioactively contaminated soil waste was generated during the decommissioning of the uranium conversion plant in KAERI, and research on the proper disposal of this waste has been actively conducted.

Numerous minerals in the soil can become glass-ceramic through the phase change of minerals during the sintering process. This method is effective in reducing the volume of waste and the glass-ceramic waste form has excellent mechanical strength and leaching resistance. In this study, the optimum temperature and time conditions were established for the production of glass-ceramic sintered body of soil. The compressive strength and leachability of the sintered body made by applying the optimal conditions to simulated waste was confirmed.

The basic physicochemical properties of simulated soil waste were identified by measuring the pH, moisture content, density, and organic matter content. The elemental compositions in the soil was confirmed by XRF. Soils were classified by particle size, and each sample was compressed with a pressure of 150 MPa or more to prepare a green body. Based on the TG-DSC analysis, an appropriate heating temperature was set ($>1,000^{\circ}\text{C}$), and the green body was maintained in a muffle furnace for 2~6 hours. The optimal sintering conditions were selected by measuring the compressive strength and volume reduction efficiency of the sintered body for each condition. The difference between the green body and sintered body was observed by XRD and SEM. In the experiments for evaluation of additives, the selected chemical substances were mixed with the soil sample in a rotator. Based on the results of TG-DSC, sintered body was made at 850°C , and the compressive strength and volume reduction were compared. Based on the results, the most effective additive was determined, and the appropriate ratio of the additive was found by adjusting the range of 1~5 wt%. This study was confirmed that the sintered soil waste showed sufficient stability to meet the disposal criteria and effective volume reduction for final disposal.

Keywords: Radioactive Waste, Contaminated Soil, Sintering, Leachability, Compressive Strength, Disposal

A Study of the Physical and Chemical Properties of Aerosols Generated by Laser Metal Cutting

Hyun Jin Boo¹, Hwa Jeong Han¹, Jonghyeon Kim¹, Su Hyeon Lee¹, Su Hyeon Kim¹, Byung Gi Park^{1,*},

Min-Ho Lee², Hee Kwon Ku², and Dae-Won Cho³

¹Soonchunhyang University

²FNC Technology

³Korea Institute of Machinery and Materials (KIMM)

*byunggi@sch.ac.kr

When decommissioning a nuclear power plant, the structure must be made to a disposable size. In general, the cutting process is essential when dismantling a nuclear power plant. Mainly, thermal cutting method is used to cutting metal structures. The aerosols generated during thermal cutting have a size distribution of less than 1 μm . The contaminated structures are able to generate radioactive aerosols in the decommissioning. Radioactive aerosols of 1 μm or less are deposited in the respiratory tract by workers' breathing, causing the possibility of internal exposure. Therefore, workers must be protected from the risk of exposure to radioactive aerosols. Prior knowledge of aerosols generated during metal cutting is important to ensure worker safety. In this study, the physical and chemical properties of the aerosol were evaluated by measuring the number and mass concentrations of aerosols generated when cutting SUS304 and SA508 using the laser cutting method.

High-resolution aerosol measuring equipment (HR-ELPI+, DEKATI) was used to measure the concentration of aerosols. The HR-ELPI+ is an impactor-type aerosol measuring equipment that measures the aerosol number concentration distribution in the aerodynamic diameter range of 6 nm to 10 μm in real-time. And analyze the mass concentration of the aerosol according to the diameter range through the impactor. ICP-MS was used for elemental mass concentration analysis in the aerosol. Analytical elements were Fe, Cr, Ni and Mn.

For the evaluation of physical and chemical properties, the MMAD of each element and CMAD were calculated in the aerosol distribution. Under the same cutting conditions, it was confirmed that the number concentration of aerosols generated from both materials had a uni-modal distribution with a peak around 0.1 μm . CMAD was calculated to be 0.072 μm for both SUS304 and SA508. The trend of the CMAD calculation results is the same even when the cutting conditions are changed. In the case of MMAD, it was confirmed that SUS304 had an MMAD of around 0.1 μm in size for only Fe, Cr and Mn. And SA508, Fe, Cr, Ni and Mn were all confirmed to have MMAD around 0.1 μm in size. The results of this study show that a lot of aerosols in the range of less than 1 μm , especially around 0.1 μm in size, are generated when metal is cut using laser cutting. Therefore, in order to protect the internal exposure of workers to laser metal cutting when decommissioning NPPs, it is necessary to protect from nano-sized aerosols beyond micron size.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 1520300060).

Keywords: Laser Cutting, High-resolution Aerosol Measurement, Aerodynamic Diameter, Number Concentration, Mass Concentration

5분과

제염해체(Poster)

Preliminary Mass Flow Diagram of a Thermochemical Process for Decontamination of Radioactive-combustible Waste

Min Ku Jeon^{1,*}, Sung-Wook Kim¹, Maengkyo Oh^{1,2}, Hee-Chul Yang¹, and Keunyoung Lee¹

¹Korea Atomic Energy Research Institute (KAERI)

²Yonsei University

*minku@kaeri.re.kr

The Korea government decided to shut down Kori-1 and Wolsung-1 nuclear power plants (NPPs) in 2017 and 2019, respectively, and their decommissioning plans are underway. Decommissioning of a NPP generates various types of radioactive wastes such as concrete, metal, liquid, plastic, paper, and cloth. Among the various radioactive wastes, we focused on radioactive-combustible waste due to its large amount (10,000–40,000 drums/NPP) and environmental issues. Incineration has been the traditional way to minimize volume of combustible waste, however, it is no longer available for this amount of waste. Accordingly, an alternative technique is required which can accomplish both high volume reduction and low emission of carbon dioxide.

Recently, KAERI proposed a new decontamination process for volume reduction of radioactive-combustible waste generated during operation and decommissioning of NPPs. This thermochemical process operates via serial steps of carbonization-chlorination-solidification. The key function of the thermochemical decontamination process is to selectively recover and solidify radioactive metals so that radioactivity of the decontaminated carbon meets the release criteria.

In this work, a preliminary version of mass flow diagram of the thermochemical decontamination process was established for representative wastes. Mass balance of each step was calculated based on physical and chemical properties of each constituent atoms. The mass flow diagram provides a platform to organize experimental results leading to key information of the process such as the final decontamination factor and radioactivity of each product.

Keywords: Decontamination, Combustible Waste, Carbonization, Chlorination, Solidification

Safety Study on Residual Power Transmitted From Laser-cut Object for Nuclear Decommissioning Application

Jae Sung Shin*, Ki-Hee Song, Seong Y. Oh, and Seung-Kyu Park

Korea Atomic Energy Research Institute (KAERI)

*jsshin12@kaeri.re.kr

Laser cutting has been attracting attention as a next-generation tool in application for nuclear decommissioning. It enables high-speed cutting of thick metal objects, and its narrow kerf width greatly reduces the amount of secondary waste compared to other cutting methods. In addition, it only requires the relatively small cutting head without any complicated equipment, and long-distance cutting apart from a laser generator is possible using beam delivery through optical fiber. And there is almost no reaction force because it is non-contact thermal cutting. For these reasons, the laser cutting is very advantageous for remote cutting.

In laser cutting, the irradiated laser power is absorbed and consumed to melt the material of the cutting target. When the applied laser power is greater than the power consumed for melting, the residual power is transmitted to the back of the cut object. This residual power may unintentionally cut or damage undesired objects located behind the cutting target. In order to prevent this, it is necessary to adjust the laser power for each thickness of the target object to be cut, or to increase the distance between the cut target and the surrounding structures so that the transmitted power density can be sufficiently lowered.

In this work, safety study on residual power that penetrates laser-cut objects was conducted. Experimental studies were performed to find safe conditions for irradiation power density that does not cause surface damage to the stainless steel by adjusting the laser power and stand-off distance from the target.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (No. RS-2022-00155395).

Keywords: Laser Cutting, Laser Processing, Nuclear Decommissioning, Nuclear Dismantling

Control Measures of Safety and Risks for Induction Melting Facilities of Radioactive Metal Waste

Kwan Seong Jeong^{1,*}, In-Ho Yoon¹, Keun Young Lee¹, and Jung Jun Lee²

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Institute of Nuclear Safety (KINS)

*ksjeong1@kaeri.re.kr

An induction melting facility includes several work health and safety risks. To manage the work health and safety risks, care must be taken to identify reasonably foreseeable hazards that could give rise to risks to health and safety, to eliminate risks to health and safety so far as is reasonably practicable. If it is not reasonably practicable to eliminate risks to health and safety, attention have to be given to minimize those risks so far as is reasonably practicable by implementing risk control measures according to the hierarchy of control in regulation, to ensure the control measure is, and is maintained so that it remains, effective, and to review and as necessary revise control measures implemented to maintain, so far as is reasonably practicable, a work environment that is without risks to health or safety.

The way to manage the risks associated with induction melting works is to identify hazards and find out what could cause harm from melting works, to assess risks if necessary – understand the nature of the harm that could be caused by the hazard, how serious the harm could be and the likelihood of it happening, to control risks – implement the most effective control measures that are reasonably practicable in the circumstances, and to review control measures to ensure they are working as planned.

Keywords: Induction Melting, Radioactive Metal Waste, Safety

Development of 3D Mapping Procedure for Derived Reasonable Work Plan in the Case of Decommissioning Nuclear Power Plant

Jaeryang Kim*, Sangbok Lee, Dongyoon Kim, Jungmook Kim, Chiil Ohk, Giwon Lee, Doohee Lee,

and Jungki Kang

LC Gen Co. Ltd

*jaeryang0330@naver.com

In 2017, Kori unit 1 nuclear power plant was permanently shut down at the end of its life. Currently, Historical Site Assessment (HSA) for MARSSIM characteristics evaluation is being conducted according to the NUREG-1575 procedure, this is conducted through comprehensive details such as radiological characteristics preliminary investigation and on-site interview. Thus, the decommissioning of nuclear power plant must consider safety and economic feasibility of structures and sites. For this purpose, the establishment of optimal work plan is required which simulations in various fields. This study aims to establish procedure that can form a basis for a rational decommissioning plan using the virtual nuclear power plant model.

The mapping procedure for 3D platform implementation consisted of three steps. First, scan the inside and outside of the nuclear power plant for decommissioning structure analysis, 3D modeling is performed based on the data. After that, a platform is designed to directly measure the radiation dose rate and mapped the derived to the program. Finally, mapping the radiation dose rate for each point in 3D using the radiation dose rate calculation factor according to the time change the measured value created on the 3D mapping platform.

When the mapping is completed, it is possible to manage the exposure dose of workers according to the ALARA principle through the change of radiation dose rate over time because of visualization of the color difference to the radiation dose rate at each point. For addition, the exposure dose evaluation considering the movement route and economic feasibility can be considered using developed program.

As the interest in safety accidents for workers increases, the importance of minimum radiation dose and optimal work plan for workers is becoming increasingly important. Through this mapping procedure, it will be possible to contribute to the establishment of reasonable process for dismantling nuclear power plant in the future.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (20191510301120, Development of industrial Technology & Standards for Radiological Characteristics Phase of Permanent Shutdown NPPs).

Keywords: Dismantling Plan, Radiation Exposure Dose, 3D Mapping Procedure, Structural Modeling

The Establishment Strategy of Effective Management Plan for Stakeholder Engagement During Decommissioning of Nuclear Facilities

Kwan Seong Jeong*, In-Ho Yoon, and Chang Hyun Roh

Korea Atomic Energy Research Institute (KAERI)

*ksjeong1@kaeri.re.kr

A large spectrum of possible stakeholders and important factors for safety improvement during decommissioning of nuclear facilities should be identified. Decommissioning includes additional aspects which are of interest to a wider range of stakeholders. The way in which local communities, the public in general, and a wide range of other parties are engaged in dialogue about decommissioning of nuclear facilities is likely to become an increasingly important issue as the scale of the activity grows. Timely stakeholder involvement may enhance safety and can encourage public confidence. Stakeholder engagement may result in attention to issues that otherwise might escape scrutiny. Public confidence is improved if issues that are raised by the public are taken seriously and are carefully and openly evaluated. Experience in many countries has shown that transparency can be an extremely effective tool to enhance safety performance. It sets out the development and implementation of an effective two-way process between the organization and stakeholders. Meaningful engagement is characterized through a flow of communication, opinions and proposals in both directions and the use of collaborative approaches to influence and explain decisions. The process is one in which an organization learns and improves its ability to perform meaningful stakeholder engagement while developing relationships of mutual respect, in place of one-off consultations. The evolving nature of this process is particularly relevant to pipeline projects, which will have differing stakeholder engagement requirements at each phase of the project lifecycle.

Activity undertaken at all stages of the process should be documented to ensure engagement success can be reviewed and improved and to ensure historical decisions or engagements are captured in case stakeholders change during the progression of time and previous consultation records are required.

Keywords: Decommissioning, Nuclear Facility, Safety, Stakeholder

Understanding of Scintillators for Radiation Detection Under Intense Gamma-rays Fields for Nuclear Decommissioning

Changhyun Roh, Su Jung Min, Seon Kwang Yoon, Chae Hun Lee, Sang Yoon Park, Sang Bum Hong,
and Bum Kyung Seo*

Korea Atomic Energy Research Institute (KAERI)

*bumkja@kaeri.re.kr

The safe, efficient and cost-effective decommissioning and dismantling of radioactive facilities requires the accurate characterization of the radionuclide activities and dose rate environment. And it is critical across many nuclear industries to identify and locate sources of radiation accurately and quickly. One of the more challenging aspects of dealing with radiation is that you cannot see it directly, which can result in potential exposure when working in those environments. Generally, semiconductor detectors have better energy resolution than scintillation detectors, but the maximum achievable count rates are limited by long pulse signals. Whereas some high pure germanium detectors have been developed to operate at high count rates, and these HPGe detectors could obtain gamma-ray spectra at high count rates exceeding 1 Mcps. However, HPGe detectors require cooling devices to reduce the leak currents, which becomes disadvantageous when developing portable radiation detectors. Furthermore, chemical-compound semiconductor detectors made of cadmium telluride and cadmium zinc telluride are popular, because they have good energy resolution and are available at room temperature. However, CdTe and CZT detectors develop irradiation-induced defects under intense gamma-ray fields. In this Review, we start with the fundamentals of gamma rays detection and review the recent developments in scintillators gamma-ray detectors. The key factors affecting the detector performance are summarized. We also give an outlook on the field, with emphasis on the challenges to be overcome.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) Grant funded by the Ministry of Science and ICT (RS-2022-00154985).

Keywords: Nuclear Decommissioning, Intense Gamma-ray, Radiation Detection, Scintillators

Magnetic Hierarchical Titanium Ferrocyanide for the Selective Removal of Radioactive Cesium From Water

Hee-Man Yang^{1,2,*}

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*hmyang@kaeri.re.kr

The magnetic Cs adsorbent functionalized with hierarchical titanium-ferrocyanide were fabricated for the highly efficient magnetic removal of radioactive cesium from water.

The new magnetic Cs adsorbent has a core-shell structure that comprises a Fe_3O_4 core, an interlayer of SiO_2 , and a titanium-ferrocyanide-shell with hierarchical nanostructure.

At first, the magnetic Fe_3O_4 nanoparticles synthesized via a hydrothermal reaction were coated with SiO_2 . Then, TiO_2 were coated on the surface of SiO_2 coated magnetic nanoparticles. Finally, the hierarchical titanium-ferrocyanide composed of 2-dimensional TiFC flakes was fabricated on the surface of core-shell $\text{MNP@SiO}_2\text{@TiO}_2$ microparticles using a TiO_2 sacrificial template via a simple reaction with potassium ferrocyanide (FC) based on the Kirkendall-type diffusion. The resulting magnetic Cs adsorbent shows higher adsorption capacity of 416 mg/g than other magnetic Cs adsorbents (below 200 mg/g) because of the increased effective surface area of hierarchical titanium-ferrocyanide. Therefore, our Magnetic Cs adsorbent functionalized with hierarchical titanium-ferrocyanide has excellent potential for the treatment of various ^{137}Cs -contaminated sources.

Keywords: Magnetic Adsorbent, Hierarchical Structure, Radioactive Cesium, Titanium Ferrocyanide

Surface Decontaminants Composed of Poly (Vinyl Alcohol) and Phenylboronic Acid-grafted Poly (Methyl Vinyl Ether-alt-Mono-Sodium Maleate)

Hee-Man Yang^{1,2,*}, Sung-Wook Kim¹, Jong-Yun Kim^{1,2}, and Hyung-Ju Kim¹

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*hmyang@kaeri.re.kr

Wide-area surface decontamination is essential in the emergency situation of release of radioisotopes to public such as nuclear accident or terrorist attack. Here, a self-generated hydrogel based on the reversible complex between poly (vinyl alcohol) (PVA) and phenylboronic acid-grafted poly (methyl vinyl ether-alt-mono-sodium maleate) (PBA-g-PVM-SM) was developed to remove the radioactive cesium from surface. Two aqueous polymeric solutions of PVA and PBA-g-PVM-SM containing sulfur-zeolite were simultaneously applied to surfaces, which subsequently self-generated a hydrogel based on the PBA-diol ester bond. The sulfur-zeolite suspended in hydrogel selectively remove the ¹³⁷Cs from contaminated surface and easily separated from the dissociable used hydrogel by simple water rinsing. In radioactive tests, the resulting hydrogel containing sulfur-chabazite displayed high ¹³⁷Cs removal efficiencies of 96.996% for painted cement and 63.404% for cement, which was 2.33 times higher than that of commercial strippable coating (Decongel). Considering the intrinsic various ion-exchange ability of zeolite, our hydrogel system has the excellent potential for the effective removal of various hazardous contamination including radionuclides from the surface.

Keywords: Surface Decontaminat, Radioactive Cesium, Self-generated Hydrogel, Poly (vinyl alcohol) (PVA), Phenylboronic Acid-grafted Poly (methyl vinyl ether-alt-mono-sodium maleate)

Management Prospects of Solid Radioactive Waste Generated From Fukushima Nuclear Power Plant Accident

Seok-Ju Hwang*, Sunghoon Hong, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hsjhhhhh@khnp.co.kr

Solid radioactive waste such as rubble, trimmed trees, contaminated soil, metal, concrete, used protective clothing, secondary waste, etc. are being generated due to the Fukushima nuclear power plant accident occurred on March 11, 2011. Solid radioactive waste inside of Fukushima NPP is estimated to be about 790,000 m³. The solid radioactive waste includes combustible rubble, trimmed trees, and used protective clothing, and is about 290,000 m³. These will be incinerated, reduced to about 20,000 m³ and stored in solid waste storage. The radioactive waste incinerator was completed in 2021. About 60,000 m³ of rubble containing metal and concrete with a surface dose rate of 1 mSv/h or higher will be stored without reduction treatment. Metal with a surface dose rate of 1 mSv/h or less are molten, and concrete undergoes a crushing process. About 60,000 m³ of contaminated soil (0.005 ~1 mSv/h) will be managed in solid waste storage without reduction treatment. The amount of secondary waste generated during the treatment of contaminated water is about 6,500 huge tanks, and additional research is being conducted on future treatment methods.

Keywords: Fukushima, Solid Radioactive Waste, Rubble, Contaminated Soil, Incinerator

Improving the Detection Efficiency of the Radiation Detector Through Secondary Solute Modification

Su Jung Min^{1,2}, Yong Dae Park², Bum Kyung Seo², Chang Hyun Roh², Sang Bum Hong^{2,*}, and Jae Hak Cheong¹

¹Kyung Hee University

²Korea Atomic Energy Research Institute (KAERI)

*sbhong@kaeri.re.kr

Organic scintillator is easy to manufacture a large size and the fluorescence decay time is short. However, it is not suitable for gamma measurement because it is composed of a low atomic number material. Organic scintillation detectors are widely used to check the presence or absence of radiation. The fluorescence of organic scintillators is produced by transitions between the energy levels of single molecules. In this study, an organic scintillator development study was conducted for use in gamma measurement, alternative materials for secondary solute used in basic organic scintillators were investigated, and the availability of alternative materials, detection characteristics, and neutron/gamma identification tests were performed. In other words, a secondary solute showing an improved energy transfer rate than the existing material was reported, and the performance was evaluated. 7-Diethylamino-4-methylcoumarin (DMC), selected as an alternative material, is a benzopyrone derivative in the form of colorless crystals, has high fluorescence and high quantum yield in the visible region, and has excellent light stability. In addition, it has a large Stokes shift characteristic, and solubility in solvent is good. Through this study, it was analyzed that the absorption wavelength range of DMC coincided with the emission wavelength range of PPO, which is the primary solute. Through this study, it was confirmed that the optimal concentration of DMC was 0.04wt%. As a result of performing gamma and neutron measurement tests using a DMC-based liquid scintillator, it showed good performance (FOM=1.42) compared to a commercial liquid scintillator. Therefore, the possibility of use as a secondary solute was demonstrated. Based on this, if studies on changes in the composition of secondary solute or the use of nanoparticles are conducted, it will be possible to manufacture and utilize a scintillator with improved efficiency compared to the existing scintillator.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) Grant funded by the Ministry of Science and ICT (RS-2022-00154985).

Keywords : Organic Scintillator, Fluorescence, Radiation Detection, Energy Transfer

Development of Remote Machining Technique in Hotcell for Compact Tension Specimens of Materials Harvested From Nuclear Reactor

Kisoo Heo*, Younggwan Jin, Seungje Baik, and Minchul Kim

Korea Atomic Energy Research Institute (KAERI)

*Heokisoo@kaeri.re.kr

Recently, it is being carried out the project to evaluate the properties of materials harvested from nuclear reactor after the decommissioning of Kori Unit 1. However, it is not sufficient adequate machining equipment and remote machining technique to perform the projects for evaluation of materials harvested from nuclear reactor. Thus, it is required to develop the remote machining technique in hotcell to evaluate the mechanical properties of nuclear reactor materials. The machining technique should be performed inside a hotcell to evaluate mechanical properties of materials harvested from nuclear reactor and is essential to prevent radiation exposure of workers. Also, it is essential to design the apparatus and develop the machining process so that it can be operated with a manipulator and minimize contamination in hotcell. In this research, development of remote specimen machining technique in hotcell such as machining apparatus, technique and process for compact tension specimens of material harvested from nuclear reactor are described. Remote machining technique will be useful in specimen machining to evaluate changes in mechanical properties of materials harvested in high-radioactive reactor. Also, it is expected that various types of specimens can be machining by applying the developed machining technique in the future.

Keywords: Machining, Hotcell, Materials Harvested, Nuclear Reactor, Compact Tension Specimens

Method of Preventing Radioactive Leakage of Exhaust Gas Treatment Facility

In Jun Lee^{†,*}, Ju Sakong[†], and Seoyong Choi²

[†]KEPCO Korea Plant Service & Engineering (KEPCO-KPS)

²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*injun@kps.co.kr

Withdrawal

~~Inquiry Into the Method to Reduce the Amount of Radioactive Waste~~

~~Injun Lee^{†,*}, Panjae Kim[†], Sangjun Kim[†], and Seoyong Choi²~~

~~[†]KEPCO Korea Plant Service & Engineering (KEPCO-KPS)~~

~~²Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)~~

~~*injun@kps.co.kr~~

Withdrawal

Comparison of ICRP Publication Dose Conversion Factors and Estimation of Exposure Dose for Decommissioning Workers

Jinho Son, Seongjun Kim, Wosuk Choi, Hwapyoung Kim, Changju Song, Heejin Kim, and Taeyoung Kong*

Chosun University

*tykong@chosun.ac.kr

Radiation workers receive exposure during radiation works such as decontamination or cutting of metals and concrete in decommissioning nuclear power plants. To reduce occupational exposure, various radiation protection measures should be prepared by estimating the exposure dose in advance. RESRAD-RECYCLE, the computer code, is generally used for estimating occupational dose due to handling metals contaminated with radioactive materials. However, RESRAD-RECYCLE used the dose conversion factors (DCF) of EPA FGR No. 11 based on ICRP Publications 30 and 48 published in the 1980s for internal exposure estimation.

This study compared the DCFs of RESRAD-RECYCLE with those of the relatively recently published ICRP Publications 119 and 141. In addition, the internal exposure dose was evaluated by changing the value of the DCFs of RESRAD-RECYCLE. As a result of the comparison, ICRP Publication 119 showed that the DCF values of most nuclides were significantly lowered. On the other hand, in the case of nuclides emitting gamma rays, there was generally no significant change in the value of DCFs. In addition, in the case of ^{65}Zn and ^{94}Nb , the DCF increased compared to the previous ICRP publications.

The exposure dose of the decommissioning workers of Hanul Units 1 and 3 and Hanbit Unit 4 was also calculated in this study. The expected radioactivity concentration of the steam generator chamber of each unit was used as the source term. The concentration of metal dust in the air generated during cutting was calculated and applied to evaluate the internal exposure dose. As a result of the dose evaluation, there was a difference in exposure dose up to 0.2 mSv in the scrap cutter scenario of Hanbit Unit 4, which generated a lot of dust and had a high radioactivity concentration. On the other hand, in the case of the slag worker, there was no difference in the dose because the working time was very short, and the inhalation of metal dust was small, even if the latest DCF was applied.

Acknowledgments

This research was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety using the financial resource granted by the Nuclear Safety and Security Commission of the Republic of Korea (No. 2106024-0222-SB110).

Keywords: DCF, Internal Exposure, Resrad-Recycle, Radiation Worker

Reduction Process of Transferred Arc Plasma Melting System

Ki-Man Park*

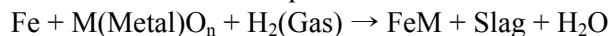
VITZRO NEXTECH

*kmpark@vitzrotech.com

This facility was developed to investigate the characteristics of metal oxide and to secure operational technology through hydrogen supply to 100 kW capacity transferred arc plasma torch and melting furnace under anoxic conditions.

Besides, the emission of pollutants generated during operation was minimized by burning the exhaust gases in the next combustion chamber and by applying a SNCR, a scrubber, etc.

The main target object was determined as a metal oxides generated as radioactive wastes when dismantling the nuclear power plant. The metal alloy was produced by supplying hydrogen during the melting process of the metal oxide. The reaction equation is as follows:



In this paper, operating conditions according to the melting temperature and hydrogen supply with iron and metal oxides were investigated, and the chemical characteristics of the alloyed metal and Slag were analyzed.

The result of this study can be used as fundamental data for the treatment and disposal of metal wastes.

Keywords: Plasma, Melting System, Radioactive Waste, Reduction Process, Disposal

Development of Plasma Melting System for the Disposal of Composite Materials

Byung-Soo Yoo*

VITZRO NEXTECH

*yoobs@vitzrotech.com

Most of the wastes generated when dismantling nuclear power plant were contaminated with low-level radioactive materials, therefore, applying a plasma melting system is a good option to dispose of the complex wastes safely.

Melting system with plasma technology was developed to dispose single metal or composite objects. Its purpose is to secure final emissions satisfying final treatment conditions by controlling oxidization/reduction reaction condition in detail during the melting process.

A hollow plasma torch applied at plasma melting system could be operated with various plasma-forming gasses such as N₂, Air, Ar, O₂, and etc.

The melting furnace was designed based on a double sealing structure to prevent risk factors; such as leaks, etc. in the reaction condition.

The effect of the external air inflow on the melting conditions was minimized by carefully designing the object input device, torch mounting part, final object discharge part, etc.

Keywords: Plasma, Melting System, Disposal, Radioactive, Composite Material

Development of Scan Simulator for Automation of Nuclear Facility 3D Modeling

Sungmoon Joo*

Korea Atomic Energy Research Institute (KAERI)

*smjoo@kaeri.re.kr

During the decommissioning of nuclear facilities, 3D digital model that precisely describes the work environment can expedite the accomplishment of the work. Thus, the workers' exposure to radiation is minimized and the safety risk to the workers is reduced, while precluding inadvertent effects on the environment.

However, it is common that the 3D model does not exist for legacy nuclear facilities as most of the initial design drawings are 2D drawings and even some of the 2D drawings are missing. Even in the case that all of the 2D drawings are intact, these initial design drawings need to be updated using as-built data because facilities get modified through years of operation. In those cases, 3D scanning can be a good option to quickly and accurately generate a structure's actual 3D geometric information. 3D scanning is a technique used to capture the shape of an object in the form of point cloud. Point cloud is a collection of large number of points on the external surfaces of objects measured by 3D scanners.

The conversion of point cloud to 3D digital model is a labor-intensive process as a human worker needs to recognize objects in the point cloud and convert the objects into 3D model, even though some of the conversion process can be automated by using commercial software packages. With the aim of full automation of scan-to-3D-model process, deep learning techniques that take point cloud as input and generate corresponding 3D model have been studied recently.

This paper introduces an efficient scan simulation method. The simulator generates synthetic point cloud data used to train deep learning models for classifying reactor parts in robotic nuclear decommissioning system. The simulator is built by implementing a ray-casting mechanism using a python library called 'Pycaster'. In order to improve the speed of simulation, multiprocessing is applied. This paper describes the ray casting simulation mechanism and compares the in-house scan simulator with an open source sensor simulation package called Blensor.

Acknowledgements

NRF-2022M2E9A2064290.

Keywords: Robotic Decommissioning, Scan, Point Cloud, Simulator

An Approach to Enhancement of Underwater Laser Cutting Process Monitoring Video

Sungmoon Joo*, Dongjun Hyun, Ikjune Kim, Jaehyun Ha, Jonghwan Lee, and Taeyoung Ko

Korea Atomic Energy Research Institute (KAERI)

*smjoo@kaeri.re.kr

For highly contaminated elements such as reactor pressure vessels or reactor internals, it is a viable option to cool-down and dismantle these elements in submerged (e.g. underwater) state. Several tools and processes such as saw cutting, water jet cutting or plasma cutting are currently used for underwater cutting, with each of them having their own advantages and disadvantages.

The main disadvantage of these existing methods, especially saw and water jet cutting, is the generation of secondary waste that then needs to be filtered out of the water. In addition, in the case of water jet cutting, a considerable amount of abrasive material is added, which must also be stored. To overcome this drawback, the feasibility of using laser cutting under water to minimize secondary waste production has been actively studied recently. One of the challenges with the underwater laser cutting is to visually monitor the cutting process. Flowing air bubbles generated by the cutting gas and the flashing light emitted from the laser and melting material prohibit visual monitoring of the cutting process.

This study introduces a method to enhance the video from a monitoring camera. Air bubbles can be detected by computing optical flows and the video quality can be enhanced by selective removal of the detected bubbles. In addition, suppressing the frame image update from flashing light area can also effectively enhance the video quality. This paper describes the simple yet effective video quality enhancement method and reports preliminary results.

Keywords: Underwater Laser Cutting, Visual Monitoring, Optical Flow, Video Enhancement

Analysis of the Foreign Cases of Building Reuse Safety Assessment After Decommissioning of Nuclear Facilities

Min Seong Kim, Ji Woo Kim, Yong Ho Jin, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

In general, after the decommissioning of nuclear facilities, buildings on the site can be demolished or reused. The NSSC (Nuclear Safety and Security Commission) Notice No. 2021-11 suggests that when reusing the building on the decommissioning site, a safety assessment should be performed to confirm the effect of residual radioactivity. However, in Korea, there are currently no decommissioning experiences of nuclear power plants, and the experiences of building reuse safety assessment are also insufficient. Therefore, in this study, we analyzed the foreign cases of building reuse safety assessment after decommissioning of nuclear facilities.

In this study, we investigated the Yankee Rowe nuclear power plant, Rancho Seco nuclear power plant, and Hematite fuel cycle facility. For each case, the source term, exposure scenario, exposure pathway, input parameter, and building DCGLs were analyzed. In the case of source term, each facility selected 9~26 radionuclides according to the characteristics of facilities. In the case of exposure scenario, building occupancy scenario which individuals occupy in reusing buildings was selected for all cases. Additionally, Rancho Seco also selected building renovation scenario for maintenance of building. All facilities selected 5 exposure pathways, 1) external exposure directly from a source, 2) external exposure by air submersion, 3) external exposure by deposited on the floor and wall, 4) internal exposure by inhalation, and 5) internal exposure by inadvertent ingestion. For the assessment, we used RESRAD-BUILD code for deriving building DCGLs. Input parameters are classified into building parameter, receptor parameter, and source parameter. Building parameter includes compartment height and area, receptor parameter includes indoor occupancy fraction, ingestion rate, and inhalation rate, and source parameter includes source thickness and density. The input parameters were differently selected according to the characteristics of each nuclear facility. Finally, they derived building DCGLs based on the selected source term, exposure scenario, exposure pathway, and input parameters. As a result, it was found that the maximum DCGL was 1.40×10^8 dpm/100 cm², 1.30×10^7 dpm/100 cm², and 1.41×10^9 dpm/100 cm² for Yankee Rowe nuclear power plant, Rancho Seco nuclear power plant, and Hematite fuel cycle facility, respectively.

In this study, we investigated the case of building reuse safety assessment after decommissioning of the Yankee Rowe nuclear power Plant, Rancho Seco nuclear power plant, and Hematite fuel cycle facility. Source terms, exposure scenarios, exposure pathways, input parameters, and building DCGLs were analyzed, and they were found to be different depending on the characteristics of the building. This study is expected to be used in the future building reuse safety assessment after decommissioning of domestic nuclear power plants.

This work was supported through the KoFONS using the financial resource granted by NSSC (No.1805016).

Keywords: Nuclear Facility, Decommissioning, Safety Assessment, Building Reuse

Application Study of Sequential Soil Flushing for Uranium Removal

Seeun Chang*, Hyun-Kyu Kim, Woo Shin Park, Sungbin Park, and Hongrae Jeon

Korea Atomic Energy Research Institute (KAERI)

*seeun@kaeri.re.kr

A sequential column experiment was conducted for uranium removal of excessively high or highly U-contaminated soils, simultaneously. Two pilot-scale acryl columns with a 24 cm ID and 48 cm length were uniformly packed with each U-contaminated soil (both < 2 mm, 119, and 22.4 Bq/g as initial U-238 activities). A column packed with soil contained very high U constant located first then sequentially located second columns with relatively lower U-contaminated soil. Thus the effluents which passed very high U-contaminated soil and having extremely high dissolved U concentration was directly inflowed the second columns.

Both columns initially and respectively flushed with demi water (or condensing water of air conditioner generated from radiation controlled area) to saturate and displace the air from the pore space. Elution was carried out with alkaline and acidic solutions, respectively, and sequentially.

The uranium removal efficiencies were found and a comparison was made with the pilot soil flushing experiments. During this study, a new approach to reducing acidic flushing waste which is considered the biggest defect of soil washing/flushing was established, and optimal factors were calculated to demonstrate industrial-scale uranium decontamination of soil with high uranium content.

Acknowledgements

This study was supported by a research grant from the Korea Atomic Energy Research Institute (Grant No. 521220-22).

Keywords: Soil Flushing, Sequential Flushing, Column Experiment, Uranium Extraction, Uranium Removal

Radiation Management During Full System Decontamination for NPP Decommissioning

Jong-hoa Kim*, Ji-hyun Yu, Bung-chae Lee, and Jang-soo Suh

SAE-AN Enertech Corp.

*jhkim4198@sae-tec.com

Minimizing of radiation exposure for the operating and decommissioning personnel is a key indicator for safe operation of the NPP. This is reflected in the application of the ALARA (As Low As Reasonable Achievable) principle. The main objectives of radiation management during full system decontamination for NPP decommissioning are to reduce the exposure dose, prevent contamination of the body and reduce solid radioactive waste. In order to reduce exposure of workers, the dose rate should be reduced by installing a temporary shield after evaluating the dose rate for the piping, component and decontamination equipment of the decontamination path before full system decontamination. Furthermore, unnecessary exposure to radiation should be reduced by thoroughly entering and exiting the radiation area and limiting the access to the high-radiation area except for workers or persons concerned. A telemetric dosimetry system should be as installed to remotely monitor radiation levels at different locations within the decontamination flow path. Remote monitoring of radiation fields using teledosimetry worked well in assessing process effectiveness and is highly recommended. However, care must be taken to place the detectors in appropriate locations. For the prevent of body contamination, it is necessary to install a fence using a heat-resistant waterproof sheet to prevent leakage of highly radioactive contamination water. When replacing high-dose filters and ion exchange resins, it is necessary to remotely monitor to reduce the exposure dose of workers.

Acknowledgments

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP), granted financial resources from the Ministry of Trade, Industry & Energy (Number 2019 1510301310).

Keywords: Full System Decontamination, Decommissioning, ALARA, Radiation Exposure

Considerations for NPP Decommissioning Project Risk Management

Gi-Lim Kim*, Hyung-Woo Seo, Sanghyun Lee, Jihwan Yu, and Hyein Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*todaylover77@khnp.co.kr

The decommissioning project of NPP is a large-scale project, with various risks. Successful implementation of the project requires appropriate identification and management of risks. IAEA considered risk management “To maximize opportunities and to minimize threats by providing a framework to control risk at all levels in the organization”. Framework-based risk management allows project managers to identify key areas in which action should be taken at an appropriate time. Also, it enables effective management of projects by supporting decision-making on sub-uncertainty. Risk could be categorized according to the source of the risk. This is called Risk Breakdown Structure (RBS), and is documented as a risk assumption register through a risk identification process.

IAEA considers various factors when defining risks in accordance with ISO 31000:2009. IAEA SRS No.97 presents a recommended risk management methodology for the strategy and execution stage of the decommissioning project of nuclear facilities through the DRiMa project conducted from 2012 to 2015. The risk breakdown structure classified in DRiMa project is as follows: (1) Initial condition of facility, (2) End state of decommissioning project, (3) Management of waste and materials, (4) Organization and human resources, (5) Finance, (6) Interfaces with contractors and suppliers, (7) Strategy and technology, (8) Legal and regulatory framework, (9) Safety, and (10) Interested parties. They have various prompts for each category.

Such a strategy for dealing with risks has negative risks (threats) or positive risks (opportunities). The negative risks are as shown in avoid, transfer, mitigate and accept. On the other side, the positive risks are as shown in exploit, share, enhance and accept. During the decommissioning, a contingency infrastructure is needed to decrease the probability of unexpected events caused by negative risks. The contingency infrastructure of decommissioning project includes organization, funding, planning, legislation & regulations, information, training, stakeholder involvement, and modifications to existing programs. Since all nuclear facilities have different environmental, physical or contamination conditions, risks and treatment strategies should also be applied differently. This risk management process is expected to proceed at the stage of establishing and implementing a detailed plan for the decommissioning project of each individual plant.

Keywords: Decommissioning, Risk Management, Project Management

A Case Study on Methodology of Dose Calculation for the Embedded Pipe of Decommissioning NPP

Jihwan Yu*, Minchul Kim, and Gi-lim Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*liu20893283@khnp.co.kr

A significant amount of piping is embedded in nuclear power plants (NPPs). In decommissioning these materials must be removed and cleaned. It can then be evaluated for radioactivity content below the release level. MARSSIM presents Derived Concentration Guideline Levels (DCGLs) that meet release guidelines. Calculating DCGL requires scenarios for the placement of embedded pipe and its long-term potential location or use. Some NPPs choose to keep the embedded pipes in the building. Because others will dismantle the building and dispose of the piping in-situ, determining the disposal option for embedded piping requires the use of measurement techniques with the sensitivity and accuracy necessary to measure the level of radioactive contamination of embedded piping and meet DCGL guidelines.

The main measuring detectors used in NPPs are gas counters that are remotely controlled as they move along the inside of the pipe. The Geiger-Mueller (GM) detector is a detector commonly used in the nuclear field. Typically, this GM detector used 3-detectors that cover the entire perimeter of the pipe and are positioned at 120-degrees to each other. This is called a pipe crawler. It is very insensitive to gamma and X-ray, only measures beta-emitter and does not provide nuclide identification. The second method is a method using a high-resolution gamma-ray detector. Although not yet commercialized in many places, embedded piping is a scanning method. The technique only detects gamma-emitting nuclides, but some nuclides can be identified. Gamma-ray scanning identifies the average concentration per pipe length by the detector collimator. It is considerably longer than a pipe crawler. In addition, several techniques, including direct measurement of dose rate and radiochemical analysis after scraping sampling, are used and they must be used complementary to each other to determine the source term. Expensive sampling and radiochemical analysis can be reduced if these detectors are used to measure the radioactivity profile and to perform waste classification using scaling factor. In the actual Trojan NPP, a pipe crawler detector was used to survey the activity profile in a 26 foot of an embedded pipe. These results indicate that the geometric averaging of the factors and a dispersion values for each nuclide are constant within the accuracy factors. However, in order to accurately use the scaling factor in waste classification, it must have sample representativeness. Whether the sample through smear or scraping is representative of the radionuclide mixture in the pipe. Since the concentration varies according to the thickness of the deposit and depending on the location of the junction or bend, a lot of data are needed to confirm the reliability of the nuclide mixture.

In this study, the reliability of the scaling factor, sampling representativeness and concentration measurement accuracy problems for waste classification in decommissioning NPP were evaluated and various techniques for measuring radioactive contamination on the inner surface of embedded pipes were surveyed and described. In addition, the advantages and limitations of detectors used to measure radioactivity concentrations in embedded piping are described. If this is used, it is expected that it will be helpful in determining the source term of the pipe embedded in the NPPs.

Keywords: Decommissioning, Embedded Pipe, Pipe Crawler, Scaling Factor

A Case Study on the Remediation of Embedded Piping of Trojan Decommissioning NPP

Jihwan Yu*, Hyung-woo Seo, Gi-lim Kim, and Hyein Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*liu20893283@khnp.co.kr

Trojan Nuclear Power Plant (NPP), a four-loop PWR designed by Westinghouse and owned by Portland General Electric (PGE), reached its initial threshold in 1975 and was operational until November 1992. PGE received a Possession Only License from the NRC in May 1993. In 1995, limited decommissioning activities began at the Trojan, including the completion of a large components removal project to remove and dispose of four steam generators and pressurizers from the containment building. In April 1996, the NRC approved a plan to dismantling the Trojan NPP and began more aggressive component removal activities. At the end of 1998, part of the radioactive drainage system began to be removed, and embedded piping decontamination and survey activities began. Trojan NPP has more than 8,840 m of contaminated pipelines throughout the power block. Most of Trojan NPP's contaminated embedded piping can generally be divided into four categories drainage piping, ventilation ducts, buried process piping, and other items. For the Trojan NPP, the complete removal of contaminated and embedded piping without damaging the building would have significantly increased costs due to the structural considerations of the building and the depth of the embedded pipe. Therefore, Trojan NPP has chosen to conduct the Embedded Pipe Remediation Project (EPRP) to clean and in situ survey of most of the embedded piping to meet the Final Site Survey (FSS) acceptance criteria, with much success.

This study provides a discussion of EPRP activities in the Trojan NPP, including classification and characterization of affected piping, modeling of proposed contamination acceptance criteria, and evaluation of various decontamination and survey techniques. It describes the decontamination tools, techniques, and survey equipment and the condition of work and cost estimate costs used in these projects. To identify embedded piping and drains at the Trojan NPP, based on frequent site surveys, plan sketches showing an overview of system flow paths and connections and database were developed to identify drain inputs and headers. This approach effort has been a successful method of remediation and site survey activities. The developed database was a valuable asset to the EPRP and a Work Breakdown Structure (WBS) code was assigned to each drains and headers, allowing the embedded piping to be integrated into the decommissioning cost estimation software (Decon. Expert) and schedule, which aided in decommissioning cost estimation. Also, regular database updates made it easy to check the status of the decommissioning project data.

The waste system drain at Trojan NPP was heavily contaminated. The goal of the remediation effort is to completely remove all removable contamination and to reduce the fixed contamination below the decided contamination acceptance criteria. Accordingly, Hydrolysis, Media blast, Chemical decontamination and Pipe removal were considered as remediation option. Trojan NPP's drainage pipe decontamination option did not cause a significant corrosion layer inside the pipe and media blast was chosen as the main method for stainless steel pipe.

In particular, the decommissioning owner decontaminates most of the embedded piping in-situ to meet the FSS acceptance criteria for economic feasibility in Trojan NPP. The remaining pipe was filled with grout to prevent leaching and spreading of contamination inside the pipe. In-situ decontamination and survey of most of these contaminated pipes are considered the most cost-effective option.

Keywords: Decommissioning, Embedded Pipe Remediation Project (EPRP), Embedded pipe

Study on Thermal Stability and Structural Changes of the Sr-exchanged Natrolite

Junhyuck Im¹, Jaewoo Jung², Kiho Yang³, Donghoon Seung⁴, and Yongmoon Lee^{3,*}

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Institute of Ocean Science & Technology (KIOST)

³Chonnam National University

⁴Pusan National University (PNU)

*lym1229@pusan.ac.kr

The radioactive Sr-90, which is formed from beta decay, is well known as one of the most commonly detected nuclides in radioactive waste. In 2015, it was reported that Sr-90 was observed in some soil and metal wastes among the 516 drums of radioactive waste transferred from the decommissioning site of the Korea Research Reactor (in Seoul) to the disposal site (in Gyeongju). Decontamination and sequestration of radionuclides, including Sr, from nuclear waste is important because they are hazardous and harmful to the ecological environment. Immobilization of these nuclides using a zeolite framework is suitable and simple method that has been widely studied. Therefore, it is still necessary to continuously explore the thermal stability of various zeolites and environmental changes around adsorbed cations in zeolite pore for effective immobilization of these radionuclides. In this study, we observed the thermal stability in fully Sr-exchanged natrolite (Sr-NAT), one of small-pore zeolite, from room temperature to 350°C using the in-situ synchrotron X-ray powder diffraction and thermogravimetric (TGA) analysis. In addition, we investigated the structural changes in Sr-NAT during temperature increase by Rietveld analysis. Sr-NAT exhibited apparent zero thermal expansions (ZTE) with the thermal expansion coefficients of $-3(1) \times 10^{-6}$ at the initial stage of increasing the temperature due to dehydration process. In the section from 250°C to 300°C, a phenomenon like negative thermal expansion (NTE) occurs in which the unit cell volume of Sr-NAT decreases despite the increase in temperature. Sr-NAT maintained well its crystallinity up to 350°C, and it became amorphous at 350°C. In this study, we provide a fundamental understanding of the structural changes and thermal stability mechanism of Sr-exchanged zeolite natrolite with increasing temperature.

Keywords: Radionuclide, Radioactive Waste, Small-pore Zeolite, Thermal Expansion, X-ray Powder Diffraction

Feasibility Study on High Count Rate Application of a LaBr₃ Detector With a High Speed Digitizer for Nuclear Facility Decommissioning

Chaehun Lee^{1,*}, Seonkwang Yoon^{1,2}, Sang-bum Hong¹, and Bumkyung Seo¹

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*chlee80@kaeri.re.kr

Radiological characterization is important in decommissioning and dismantling of nuclear facilities, in order to assess the radioactivity concentration, classify the wastes, and secure workers' safety. The Some components such as Reactor Pressure Vessel (RPV) in nuclear facilities has dose rate higher than Sv/hr, thus in-situ gamma spectroscopy systems suffer from a very high count rate which causes energy resolution degradation, photo-peak shift, and count loss by pile-up and dead-time. The system must be operated in a very high count rate, in order to measure spectra precisely and to quantify radionuclide contents. In order to apply in-situ measurement in high radiation dose rate environment, the sensor, front-end electronics, and data acquisition (DAQ) should be carefully selected and designed as well as precise design of collimators and radiation shield. In this paper, the components of the detector system were selected and performance was evaluated in a high count rate before design the collimator and shield. A LaBr₃ coupled with a PMT having short decay time constant (16 nsec) was selected for high count rate application, and two different amplifiers (a conventional charge sensitive preamplifier with 50 usec decay time constant, and wide-band voltage amplifier) were tested. As DAQs, DT5781 (14 bit, 100 MS/s, CAEN) of Pulse Height Analysis (PHA) which is conventionally used signal processing method in the gamma spectroscopy, and DT5730 (14 bit, 500MS/s, CAEN) of Pulse Shape Discrimination (PSD) which is similar to Charge to Digital Convertor (QDC) were used. The number of photons incident to the detector was varied by changing the detector-source distance with Certificate Radiation Material (CRM), and compared to the output count rate. The count rate capability, and energy resolution with different amplifier and DAQ was evaluated. Additionally, the performance of DAQs in extremely high count rate was evaluated with signal data generated by the emulator which can simulate the detector signal waveforms fed into the DAQ based on the measured spectrum.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) Grant funded by the Ministry of Science and ICT (RS-2022-00154985, and NRF-2020M2C9A1068162).

Keywords: High Count Rate, In-situ Measurement, Pulse Height Analysis, Pulse Shape Discrimination, Pile-up, Dead-time

Radioactivity Calculation Considering Plant-specific Operation History for Decommissioning of PWR Nuclear Power Plant

Youngjae Maeng*, Kyungsik Kim, Mijoung Lim, and Choonsung Yoo

Korea Reactor Integrity Surveillance Technology (KRIST)

*maeng@krist.co.kr

Reliable evaluation of radioactivity inventory for the nuclear power plant components and residual materials is very important for decontamination and decommissioning. This can make it possible to define optimum dismantling approaches, to determine radioactive waste management strategies, and to estimate the project costs reasonably.

To calculate radioactivity of the nuclear power plant structure, various information such as interest nuclide, cross-section, decay constant, irradiation time, neutron flux, and so on is required. Especially irradiation time and neutron flux level are very changeable due to cycle specific fuel loading pattern, the plant overhaul, cycle length. However most of the radioactivity calculations have generally been performed assuming one representative or average neutron flux during the lifetime of the nuclear power plant. This assumption may include excessive conservatism because the radioactivity level has the characteristics of saturation and decay. Therefore, considering these variables as realistically as possible could prevent overestimation.

In order to perform realistic radioactivity calculation, we developed monthly relative power contribution factor applying plant-specific operation history and cycle-specific neutron flux. The factors were applied to the radioactivity calculation. The calculation results were compared with measured values of the neutron monitors that were actually installed and withdrawn from the nuclear power plant. As a result of the comparisons, there are good agreements between the calculated values and measured values.

These accurate calculation results of radioactivity could contribute to the establishment of radioactive waste dismantling strategies, the classification of radioactive waste, and the deposit of disposal costs for safe and reasonable decommissioning of the nuclear power plant.

Keywords: Activity, Radioactive Waste, Decommissioning, Operation History, Neutron

Proposal of Evaluation Matrix for Clearance Waste Disposal Plan

Kunsu Lim*, Chang-Lak Kim, and Sanghwa Shin

Kepeco International Nuclear Graduate School (KINGS)

*etetet2005@naver.com

The Korean administration assumed that the amount of low and medium level waste generated during the decommissioning of nuclear facilities in Korea was 14,500 drums (based on 200 L) and designed the LILW repository accordingly. Accordingly, it is necessary to separate the nuclear power plant decommissioning waste into clearance waste by mobilizing means such as decontamination and cutting as much as possible, and to deregulate it together with non-radioactive waste. As a result, clearance waste and non-radioactive waste are dominated by concrete and metal, and it is necessary to evaluate how to recycle them. Many existing studies have conducted research on each recycling method, and accordingly, it can be judged that the technological maturity is sufficient. Accordingly, we would like to propose a method for comprehensive management and evaluation of concrete.

By applying the decision matrix proposed in IAEA TRS No. 401, it will be possible to compare the 5 factors (cost, technical feasibility, risk, availability of disposal, and full cycle impact). However, in the case of concrete, if the existing construction waste recycling methodologies are fully used, the technical feasibility can be considered equal. Therefore, it was judged that it would be good to introduce the aspect of public acceptance as an evaluation item instead of technical feasibility.

The amount of waste that can be generated when decommission a nuclear power plant is only insignificant compared to the total amount of waste concrete that is generated during the year. Accordingly, one option is to fully integrate the waste concrete recycling system and utilize it for road construction. Next, it is possible to suggest the option of recycling in the construction of shields in the nuclear industry, as suggested in previous studies, and the method of using it as a backfill material such as for a decommissioned NPP site or other sites.

As an example, and a draft stage, this study was evaluated based on existing studies after all options were equally weighted. When the profit and loss was evaluated in a way that a maximum of 5 points were given to each option, the case of using it as a backfill in various applications was evaluated as the best option. Unlimited recycling, such as road construction, was evaluated to be highly damaging in terms of public acceptance.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords : Decommissioning, Concrete, Clearance Waste, Recycle

A Review of Remediation Actions and Applicable Scenarios to be Considered at the Time of License Termination for NPPs Decommissioning

Hyung-woo Seo*, Ji-hwan Yu, Gi-lim Kim, and Hyein Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*sehywo1727@khnp.co.kr

The goal of the decommissioning of nuclear facilities is to remove the regulations from the Nuclear Safety Act. The media that can be considered at the time of remediation stage may usually include soils, buildings, and underground materials. In addition, underground materials may largely be the groundwater, buried pipes, and concrete structures. In fact, it can be seen that calculations of the Derived Concentration Guideline Level (DCGL) and ALARA action levels was conducted in the case of overseas decommissioning experiences of Nuclear Power Plants (NPPs). Therefore, the aim of this study is to review the remediation activities and scenarios applied for the calculation of ALARA action level from the overseas decommissioned nuclear power plants.

Media that can be considered for DCGL calculation at the time of license termination may differ from site to site. If the DCGL for the target media was derived, whether additional remediation actions are required under the DCGL value from the ALARA perspective was identified by calculating the ALARA action levels in the case of the U.S. The activities to determine whether additional clean-up is justified under the regulatory criteria are remediation actions which is dependent on the material contaminated. Therefore, the typical materials that can be subjected to remediation are soils and structure basements in the overseas cases. Remediation actions involved in the decommissioning process on the structure surfaces can be typically considered to be scabbling, shaving, needle guns, chipping, sponge and abrasive blasting, pressure washing, washing and wiping, grit blasting, and removal of contaminated concrete.

For the cost-benefit analysis of the media subject to DCGL calculation, it is necessary to assume a scenario for the remediation actions of the target media. The scenarios can be largely divided into two types. Those are basement fill and building occupancy scenario. In basement fill mode, buildings and structures on the site are removed, and the effect of receptors from the contamination of the remaining structures is considered. In the building occupancy mode, it is assumed that the standing building remains on the site after the remediation stage. It is a situation to evaluate how the effect of additional remediation actions changes as the receptors occupy inside of the contaminated building. Therefore, parameters such as population density, area being evaluated, monetary discount rate, numbers of years, etc. can be set and assessed according to the scenarios.

Keywords: Decommissioning, Remediation Actions, Reuse Scenario, DCGL

Evaluation of Dose to Public Due to Hypothetical Events During Dismantling of Steam Generator

Kyeong Ju Lee* and Chang Lak Kim

Kepco International Nuclear Graduate School (KINGS)

*xelloss606@gmail.com

Kori Unit 1, Korea's first commercial nuclear power plant is preparing to dismantle after 40 years of power supply. However, unlike the public dose assessment for nuclear power plants in operation, the dose assessment for the public due to abnormal events during the decommissioning of nuclear power plants is insufficient. Therefore, in this study, the steam generator chamber is selected as hypothetical events target among metal waste, which is a major radioactive material generated during the decommissioning of nuclear power plant. In addition, the possible abnormal event scenarios and effective dose to public in the Exclusion Area Boundary due to the released radioactive materials are predicted during the disassembly and transportation of the steam generator.

For the source term that can be released during the dismantling of the steam generator, the inventory of each radionuclide is evaluated based on the smear test results of the steam generator replaced in Kori Unit 1 in 1998. To evaluate the diffusion of radioactive material, the atmospheric dispersion factor (χ/Q , sec/m³) is calculated through the PAVAN code of the US NRC using the meteorological data of the Kori nuclear power plant for 3 years from 2019 to 2021 according to IAEA recommendations. For the assessment of the public dose, the external dose coefficient and inhalation coefficient of the ICRP and the inhalation rate of the NRC Regulatory Guide 1.3 are referred.

It is confirmed that the effective dose to the public in the Exclusion Area Boundary due to the abnormal event during the dismantling of the steam generator is much lower than the effective dose standard value of 250 mSv for 2 hours after the event in the Exclusion Area Boundary.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP Project No. 20204010600130) and the Ministry of Trade, Industry and Energy (MOTIE) of the Republic of Korea.

Keywords: Dismantling, Abnormal Event Scenarios, Atmospheric Dispersion Factor, Effective Dose

Computational Model Development of a Combined Electrolysis and Catalytic Exchange (CECE) Process for Heavy Water Pre-enrichment and Detritiation

Kwang-Rag Kim*, Jong-Yun Kim, Sang-Ho Lim, and Do-Hee Ahn

Korea Atomic Energy Research Institute (KAERI)

*krkim1@kaeri.re.kr

A simulation model was developed for heavy water pre-enrichment and detritiation by the Combined Electrolysis and Catalytic Exchange (CECE) process.

In the CECE process, heavy water enrichment and detritiation are based on the principle that concentrated in to water phase through an isotopic exchange reaction between water vapor and hydrogen gas produced by a water electrolysis.

An operational analysis for a liquid phase catalytic exchange column was carried out by the model equations, composed of a material balance and combined equilibrium relationships for a scrubbing and catalyst bed, respectively.

As a result of simulation, the optimum flow ratio of water to the rising hydrogen gas in contact with the down-coming water was predicted as the key variables in the separation performance analysis at a given feed flow rate and isotopic composition.

From a graphical approach based on this model, the operating conditions can be determined within the range where the operating line does not meet the combined equilibrium line before reaching the specified target concentration.

Keywords: Heavy Water, Combined Electrolysis and Catalytic Exchange Process (CECE), Enrichment, Detritiation

Review of Transport System for Decommissioning Waste Containers

Woo-Yong Kim*, Young Oh Lee, Tae Su Kim, and Moonoh Kim

KONES Corp.

*kwy@kones21.com

In preparation for the decommissioning of Kori unit 1 of the nuclear power plant (NPP), new containers of package, transportation, and disposal are being developed that reflect the type, generation amount, and radiological characteristics of decommissioning waste. The containers under development have internal volumes of $1 \text{ m}^3 \sim 14 \text{ m}^3$ and loading weights of 1 ton $\sim$ 35 tons, which are larger in size and have a higher loadable weight compared to the 200 L drum and IP-2 type transport container currently being used for packaging and transporting waste. So, there is a limit to handling new containers using existing transport systems (cranes, spreaders, forklifts, transport vehicles, etc.).

Therefore, in this study, the status of handling equipment in NPP and disposal facilities was reviewed, the flow from the generation to disposal of decommissioning waste was analyzed, and the possibility of handling new container or the necessity of introducing new systems were derived.

Except for some high-dose/high-radioactive wastes among decommissioning wastes, all wastes are finally disposed of through decommissioning area, temporary storage facility, waste treatment facility, waste storage facility, and receipt and storage building. The decommissioning area, temporary storage facility, and waste treatment facility are newly established areas for the decommissioning and should be equipped with a spreader/crane with a lifting weight of 15 tons, 35 tons, and 40 tons in consideration of the weight of the package to be handled in the zone. The waste storage facility has a 7.5 tons crane, so it can handle only some of the lower weight of the 5 to 35 tons package that is expected to be handled. Therefore, additional installation of spreaders/cranes, each with a lifting capacity of 15 tons and 40 tons, is required.

The maximum loading weight of forklifts and transport vehicles operating at NPP, and disposal facilities is 10 tons and 12.6 tons, respectively. To transport the package, the facility must additionally install 15 tons and 40 tons forklifts, and 40 tons transport vehicles. Since the lifting weight of the crane installed on the transport vessel is also low at 12.5 tons, it is necessary to change the design of the existing or replace it with 40 tons to handle high-weight package.

The results of this study will be used as basic data for the establishment of transport systems in the relevant area and facility, and design requirements for each equipment will be derived through additional research.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No.20211510100040).

Keywords: NPP Decommissioning, Radioactive Waste, Packaging and Container, Transport Equipment

A Study of Dismantling Steam Generators for Kori Unit 1, 2

Geunyoung Goh* and Chang-Lak Kim

Kepeco International Nuclear Graduate School (KINGS)

*kohkeun12@gmail.com

Reactor pressure vessels and steam generators generated in the process of dismantling nuclear power plants or replaced steam have various shape and occupy a considerable amount of the disposal site when disposed of in original shape. For the development of domestic technologies related to the disposal of large wastes, it is necessary to secure technologies for reducing large radioactive metal wastes, including technologies such as decontamination, cutting, melting, and residual radioactivity evaluation.

Cases of disposal of steam generators in Europe and the United States were investigated. Except for u-tubes, steam generators are less contaminated or easily decontaminated, so it is possible to reduce the volume of waste subject to final disposal by exempting a significant amount through appropriate treatment.

Korea Hydro & Nuclear Power Co. is currently temporarily storing 24 steam generators at 41.6 billion won. This paper presents a method to exempt more parts of the steam generator and reduce the volume of waste by properly combining mechanical cutting thermal cutting and melting to dispose of the steam generator.

Currently the decontamination and dismantling industries of nuclear facilities are gradually expanding around the world. Therefore, it is necessary to localize the treatment technology for metal waste generated during maintenance and dismantling. The result of this study can be used to establish waste reduction and disposal method for dismantling steam generators.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea.

Keywords: Decontamination, Steam Generators, U-tubes, Dismantling

Dose Evaluation on the Post-closure Scenario of Self-disposal Landfill for Decommissioning Metal/Concrete Wastes

Jaewon Park*, Juyub Kim, and Hyungi Byun

FNC Technology

*jpark@fnctech.com

Decommissioning of a nuclear power plant (NPP) generate large amounts of various types of wastes. In accordance with the Nuclear Safety and Security Commission Notice of Korea (No. 2020-6), they are classified as High Level Waste (HLW), Intermediate Level Waste (ILW), Low Level Waste (LLW), Very Low Level Waste (VLLW) and Exempt Waste (EW) according to specific activities. More than 90% of the wastes are at exempt level, mostly metal and concrete wastes with low radioactivity, of which the concentrations of nuclides is less than the allowable concentration of self-disposal. The self-disposal or recycling of these wastes is widely used worldwide.

More than 10,000 drums, based on 200 L drum, are expected to be produced in the decommissioning process of a unit of nuclear power plant. Due to the limited storage capacity of the intermediate & low level waste disposal facility in Gyeongju, recycling and self-disposal of EW are actively recommended in Korea.

A variety of scenarios were proposed for recycling and self-disposal of decommissioning metal/concrete wastes, and a computational program called REDISA was developed to perform the dose evaluation for each recycling and self-disposal scenario. The REDISA computer program can calculate external and internal exposure doses by simulating the exposure pathways from waste generation, thru transport, processing, manufacture, to the final destination of recycling or self-disposal.

In this study, the self-disposal scenario was only considered for the dose evaluation. Many studies have been conducted to evaluate the exposure doses of the radioactive waste disposal sites. However, there have been few researches on dose evaluation for self-disposal landfills.

In particular, the dose evaluation is important not only during the operation period, but also for a long period after the facility is closed. To this end, we developed a conceptual model for dose evaluation for post-closure scenarios of the self-disposal landfill of decommissioning metal/concrete wastes with reference to the methodology of IAEA-TECDOC-1380.

The model incorporates three exposure pathways, including external exposure from contaminated soil, internal exposure by inhalation, and internal exposure by ingestion of water and food grown in contaminated soil. The duration of the dose evaluation is set to 100,000 years after the closure of landfill facility. Co-60 was selected as dominant nuclide, and dose evaluation was performed based on unit specific activity of 1 Bq/g. Exposure doses shall be verified for their application in accordance with the annual dose limit of 10 μ Sv/yr for self-disposal. As a result, the post-closure scenario of self-disposal landfills have shown negligible effects on public health, which means that the exposures doses from transportation and operational processes should be considered more carefully for self-disposal of decommissioning metal/concrete wastes.

Acknowledgements

This work was supported by the Energy Efficiency & Resources of the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No. 20203210100430).

Keywords: Decommissioning, Decommissioning Waste, Dose Evaluation, Self-disposal Landfill

Case Study of Decommissioning Waste Treatment Method of Zion Nuclear Power Station

Cheong Won Lee*

Korea Atomic Energy Research Institute (KAERI)

*cwlee2020@kaeri.re.kr

Kori unit 1 was permanently shut down in 2007 and is currently awaiting approval for decommissioning and dismantling (D&D). The wastes generated during decommissioning is estimated to be approximately 14,500 of 200 L drums. In this study, the treatment process of decommissioning wastes will be reviewed through the case of the US Zion nuclear power station (ZNPS). Zion unit 1 and 2 received an operating license in 1973 and were permanently shut down and the spent nuclear fuel was transferred to the pool in 1998. The decommissioning was carried out according to the following five steps; (1) safe storage (SAFSTOR) dormancy, (2) preparation for decommissioning, (3) establishment of independent spent fuel storage installation (ISFSI) and transfer of the spent fuel and greater than class C radioactive materials, (4) decommissioning operations and (5) site restoration. The total volume of waste generated during decommissioning was expected to be approximately $1.7 \times 10^5 \text{ m}^3$. This is far above the Kori unit 1 waste estimation because ZNPS has a history of accidents and includes soil waste. Wastes were treated differently according to their properties and locations.

Keywords: Zion, Kori, Decommissioning, Dismantling, Waste Treatment

Activated Carbon Waste Pretreatment System for Radioactive Waste Up-cycling

Kirak Lee^{1,*}, Hwan-Seo Park¹, Geun-il Park¹, Sang Won Shin², and Dong Chul Jang²

¹Korea Atomic Energy Research Institute (KAERI)

²HANIL Nuclear

*kirakki@kaeri.re.kr

In NPP (nuclear power plant), boric acid is used as a neutron absorbent. So radioactive boric acid waste are generated from various waste streams such as discharge or leakage of reactor coolant water, floor drains, drainage of equipment for operation or maintenance, reactor letdown flows and etc. Depending on KHNP, 20,015 drum (200 L drum) of concentrated boric acid waste were stored in KOREA NPP until 2019. In previous study, our group suggested the waste up-cycling process synthesizing B₄C neutron absorber using boric acid waste and activated carbon waste to innovatively reduce radioactive wastes. Radioactive activated carbon waste was utilized in off gas treatment system of NPP to capture nuclide such as I-131, C-14 and H-3. Activated carbon waste is treated as low-level radioactive waste and pre-treatment system for removing nuclide from the activated carbon waste is needed to use B₄C up-cycling process.

In this study, microwave treatment system is suggested to treat the activated carbon waste. Activated carbon waste was exposed to microwave for a few minutes and temperature of the waste was dramatically increased over 400°C. Nuclide in the activated carbon waste were selectively removed from the waste without massive production of secondary off gas waste.

Keywords: Activated Carbon Waste, B₄C, Up-cycling, Pretreatment

Paraffin Separation From Paraffin Wasteform Via Evaporation and Thermal Decomposition Method

Kirak Lee*, Hyeon Jin Eom, Hwan-Seo Park, Jung-Hoon Choi, and Hyun Woo Kang

Korea Atomic Energy Research Institute (KAERI)

*kirakki@kaeri.re.kr

Boric acid-containing B-10 is used in a nuclear reactor as a coolant and absorbs thermal neutrons generated during nuclear fission in the primary circuit. Boron-containing coolant water waste is generated from maintenance, floor drain, decontamination, and reactor letdown flows. There are two options for aqueous solution waste of boric acid. One is recycling and discharge through filtration, ion exchange, and reverse osmosis. The other is immobilization after evaporation and crystallization processes. The dry powder of boric acid waste liquid can be immobilized by cement, polymer, etc. Before the mid-1990s, concentrated boric acid waste was solidified with a cement matrix. To overcome the disadvantage of low waste loading of cement waste form, a method of solidifying with paraffin was adopted. However, paraffin solids were insufficient to be disposed of as final waste. Paraffin is a kind of soft solidified material and has low compressive strength and poor leaching resistance. As a result, it was decided as an unsuitable form for disposal. In KOREA, paraffin waste form was adopted for boric acid waste treatment in the 1990s. A large amount of paraffin waste forms about 20,000 drums (200 l drum) were generated to treat boric acid waste and were stored in nuclear power sites without disposal.

In this study, we want to obtain high-purity boric acid waste by oxidizing and decomposing solid paraffin waste form through a boric acid catalytic reaction. In this reaction, paraffin is separated in the form of various by-products, which can then be treated through a liquid waste treatment device or an exhaust gas treatment device. The proper temperature for sample decomposition during the catalytic reaction was set through TGA analysis. Compositions of by-products and residues generated at each stage of the reaction could be analyzed to determine the state during the reaction. Finally, the boric acid waste powder was perfectly separated from paraffin waste form with disposable products through this pyrolysis process.

Keywords: Boric Acid Waste, Paraffin, Evaporation, Thermal Decomposition, Wasteform

Hydrogen Isotope (H, D, T) Separation Technology Development

Chan Woo Park*, Ilgook Kim, Hyung-Ju Kim, Chan Hyun Rho, Younho Shin, In-Ho Yoon, and Hee-Man Yang

Korea Atomic Energy Research Institute (KAERI)

*chanwoo@kaeri.re.kr

Hydrogen isotopes (H, D, T) separation technologies have received great interest for treatments of tritiated liquid waste produced in Fukushima. In addition, the separated deuterium and tritium can be utilized in various industries such as semiconductors and nuclear fusion as expensive and rare resources. However, separating hydrogen isotopes in gas and liquid forms still requires energy-intensive processes. To improve efficiency and performance of hydrogen isotope separation, we are developing water electrolysis, cryosorption, distillation, isotope exchange, and hydrophobic catalyst technologies. Furthermore, an analytical method is studied to evaluate the separation of hydrogen isotopes. This presentation introduces the current status of hydrogen isotope research in this research group.

Keywords: Tritium, Deuterium, Isotope, Separation

Conflict Resolution for Each Stakeholder on the Clearance Waste Recycling From the Decommissioning of Nuclear Power Plants

Dong-Jin Kim*, Ye-Jeon Lim, So-On Park, Seong-Hyeon Lee, and Byung-Sik Lee

Dankook University

*dj960930@gmail.com

The decommissioning of Kori Unit 1 is expected to generate a large amount of clearance waste. Disposing of a large amount of clearance waste is economically costly, so a recycling method has emerged. However, clearance waste recycling is expected to cause many conflicts among various stakeholders. In the previous study, possible conflicts were selected in consideration of the domestic environment and major issues. Based on this, this study classifies stakeholders involved in conflicts by group, and suggests ways to enhance understanding by stakeholder and enhance reliability.

In this study, stakeholders are classified into four groups that share the same conflicts, and each of the following measures is suggested.

- 1) Stakeholder Engagement.
- 2) Common understanding of radiation risks, dialogue between the public/recycling industry/regulatory agency.
- 3) Incentives to promote recycling clearance waste.
- 4) Reliable outlet store for recyclable clearance waste.

The above understanding enhancement measures are presented so that a solution to conflict can be smoothly derived when designing a clearance waste-related consultative body composed of interested parties in the future. As a more specific solution, measures to enhance stakeholder trust can be suggested for each understanding enhancement measure. Reliability enhancement measures are also presented so that they can be applied to each stakeholder group, and these are as follows.

- 1) Write a stakeholder engagement plan, Measures for stakeholder participation in measuring the radioactivity concentration of clearance waste.
- 2) Active use of easy-to-understand radioactivity comparison data, Expansion of information on environmental radiation dose to public, nuclear/radiation education, Held a tour event at the nuclear power plant decommissioning site, New website for clearance waste information disclosure.
- 3) Incentives for recycling industries in which the Ministry of Environment or KHNP partially bears the losses that occur when the sales rate is low.
Incentives are provided to consumers by including recyclables of clearance waste for Green Card's green consumption points.
- 4) Online outlets open for recyclable clearance waste with easy-to-understand radioactivity comparison data.

It is expected that if the above-mentioned reliability enhancement measures are used, it will be possible to secure the trust of stakeholders and reduce the gap between stakeholders in the future clearance-related consultative body.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP), granted financial resource from the Ministry of Trade, Industry and Energy (No. 20203210 100430), Republic of Korea.

Keywords: Decommissioning, Clearance Waste, Conflict Management, Stakeholder Interaction

PERT Chart Based Laser Cutting Process Control on a Digital Simulation Platform

Ikjune Kim*, Dongjun Hyun, Jonghwan Lee, Sungmoon Joo, Jaehyun Ha, and Taeyoung Ko

Korea Atomic Energy Research Institute (KAERI)

*ikjunekim@kaeri.re.kr

The RPV internal structure is a high radio activated part and has very complex geometry. Therefore, it needs to be cut remotely with an automated cutting system to minimize the worker exposures. To do so, we made up the remote laser cutting system with a laser cutter, robot manipulator and control software system and the laser cutter is moved by the robot manipulator based on the command from the control software system.

A laser cutter is required to keep the desired standoff position between the nozzle of the laser cutter and surface of the cut target model to cut properly. Moreover, in the remote cutting process, an exact time and sequence control of the air supply and the laser emission is required for the cutting quality and the process safety.

In this study, we proposed the PERT chart-based process execution and control methodology. The PERT chart is a graph which is represented by nodes and edges. The node of the PERT chart has the information about the activity details such as activity type, execution time and related device. Using the edge we make the sequence of the desired activity execution.

A PERT chart of the cutting scenario is compiled in the control software system to creates data and thread structure to operate the physical device. We built software architecture to interpret and execute the PERT chart efficiently in the digital simulation platform which enables us to use existing pre-built simulation scenario for the laser cutting process.

In addition, we have tested various laser cutting test cases in our test bed to verify the performance of our system. The test bed environment has the shape of the RPV internal structure and is placed under water.

Acknowledgements

This work was supported by the nuclear research and development program (No. 00155255) through the national research foundation (NRF) of Korea funded by the Ministry of Science and ICT.

Keywords: PERT Chart, Laser Cutting, Process Control, Digital Simulation Platform

Evaluation of Solubility of Ethanol, 2-propanol in Supercritical Carbon Dioxide for the Application of Soil Decontamination Process

Seungil Ha, Kwangheon Park*, and Jinhyun Sung

Kyung Hee University

*kpark@khu.ac.kr

Radioactive contamination of soil on the site of a nuclear facility has a characteristic that radioactive nuclides are adsorbed into the pores between soil particles, making it quite difficult to decontaminate. For this reason, research on the development of various decontamination processes is being actively conducted. In this study, among various decontamination studies, a soil decontamination process using supercritical carbon dioxide was presented. The decontamination process uses supercritical carbon dioxide as the main solvent, which has a higher penetration power than other materials. Therefore, the process consists of the process of desorbing and extracting the target radionuclides between particles of soil. However, since nuclides exist as ions in the soil, polar chelating ligand material was introduced as an additive to nonpolar supercritical carbon dioxide for smooth chemical reactions in the soil. Thereafter, from the viewpoint of improving process continuity and efficiency, an alcohol material was introduced as an auxiliary solvent for liquefaction of chelating ligand in a solid state.

Through prior research on the selection of a solvent for liquefaction of chelating ligand, ethanol and 2-propanol were finally selected based on whether the chelating ligand was dissolved. However, if the auxiliary solvent in which the chelating ligand is dissolved is to be combined with radionuclides in the soil, it must first be well dissolved in supercritical carbon dioxide, the main solvent. Therefore, in this study, the solubility of ethanol and 2-propanol in supercritical carbon dioxide was measured and the suitability was evaluated. The temperature conditions were carried out at 40°C, the same as the previously designed decontamination process, and the measurement was conducted by adjusting the pressure and volume through a syringe pump and a variable volume device. In addition, solubility was measured based on the observation of the 'cloud point' in which the image becomes cloudy and then bright.

As a result of the experiment, several solubility points were measured at a pressure of 150 bar or less. If the flow rate ratio of supercritical carbon dioxide and auxiliary solvent derived from the results is applied to the soil decontamination process, it is expected that the process efficiency will increase in the future.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT) (No. 2018M2B2B106563521).

Keywords: Supercritical CO₂, Ethanol, 2-Propanol, Solubility, Soil Decontamination, Site Remediation

Demonstration Test for Chemical Waste Decomposition & Treatment System (CWDS)

Ju-Hyeon Park, Jung-Hyun Lee, Sung-Hyun Lee, Ki-Chul Kim*, and Chung-kyu Lee

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*kichkim@kps.co.kr

In the pilot scale test, the two scale-up factors (Electric energy per order E_{EO} , Electric energy per mass E_{EM}) were conducted to design the Chemical Waste Decomposition & Treatment System (CWDS). The CWDS consist of two kind UV lamp reactors to improve the decomposition rate of oxalic acid, which are low pressure amalgam UV lamp and medium pressure UV lamp. The two reactors were connected in series, and the hydrogen peroxide is mixed through a line mixer at the front of the reactor and injected into the reactors.

The CWDS was connected with the full system decontamination equipment to purify the residual oxalic acid after chemical decontamination process. The full system decontamination equipment were included Oxidizing Agent Manufacturing System (OAMS), Chemical Injection System (CIS), RadWaste Treatment System (RWTS) to operate the Oxidation/Reduction decontamination process and purify the process water.

After decontamination process, the waste water will be cooled down into the 40°C and passed through the UV reactor at 110 gpm with hydrogen peroxide injection. The concentration of waste water is expected oxalic acid 1,700 ~ 2,000 ppm, Iron 5 ~ 20 ppm.

As a result of the CBD test in the laboratory with simulated waste liquid, the amount of Low pressure amalgam lamp UV dose required to decompose 95% of oxalic acid in 2 m² waste water was up to 1,800 mJ/cm². The amount of medium pressure lamp UV dose was up to 450 mJ/cm² at the same condition. We conducted demonstration test using 2 m² waste water after the oxidation/reduction decontamination process, the decomposition rate 95% was obtained by low pressure amalgam UV lamp and medium pressure UV lamp reactor each.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2019 1510301310).

Keywords: Chemical Waste Decomposition, UV AOP Process, Full System Decontamination

Screening of Porous Materials for Separation of Hydrogen Isotopes Based on Cryosorption

Chan Woo Park*, Hyung Ju Kim, and In Ho Yoon

Korea Atomic Energy Research Institute (KAERI)

*chanwoo@kaeri.re.kr

Withdrawal

Changes to the Design of CHEONG JEONG NURI for the Transport of Decommissioning Radioactive Waste

Sanghyeon Eom^{1,*}, Moonoh Kim², and Woo Yong Kim³

KONES Corp.

*sheom@kones21.com

To transport radioactive waste generated during the decommissioning of Kori Unit 1, transport containers of various sizes are being developed. Since these radioactive decommissioning waste transport containers are larger than the specifications of the existing IP-2 type transport containers, which are for operational radioactive waste, design of the CHEONG-JEONG-NURI needs to be changed when transporting them to disposal facility using the CHEONG-JEONG-NURI, which carries operational radioactive waste.

In this study, design changes of the CHEONG-JEONG-NURI, cargo hold modification plan for efficient loading of radioactive decommissioning waste transport containers and radioactive decommissioning waste container loading arrangement (plan) were evaluated during the design life period (year 2034).

First, as only the IP-2 type transport container with a weight of 7.5 tons and size of 1.6 m (W) × 3.4 m (L) × 1.2m (H) can be loaded in the cargo hold, if only the decommissioning radioactive waste containers are to be loaded and transported, cargo hold needs to be reinforced.

Second, when both the radioactive decommissioning waste transport container of the same size as the current operating radioactive waste transport container, and the radioactive decommissioning waste transport container of the same size as the ISO-type transport container are to be loaded in the cargo hold of the CHEONG-JEONG-NURI and transported, the overall design changes (cargo hold size and load reinforcement) are required.

Third, since the safe working load of the CHEONG-JEONG-NURI crane is 12.5-tons, it shall be replaced with a ship crane of 35-tons or more to handle the decommissioning radioactive waste container smoothly, or a gantry crane used in general port facilities shall be installed.

When replacing with a ship crane of 35-tons or more, ship buoyancy, ship stability, and ship structural safety shall be considered. The possibility of moving in all 4 directions for smooth operation, and the possibility of lifting the transport container to a position higher than the height of the CHEONG-JEONG-NURI shall be considered.

Loading and transporting all decommissioning radioactive waste containers, which are the same size as IP-2 and ISO-type transport containers, in the cargo hold of the CHEONG-JEONG-NURI is uneconomical due to the need for overall design changes (cargo size and load reinforcement, etc.). Also, delay in delivery of the operation wastes is expected due to a long-term design change period.

Therefore, it is considered reasonable to load and transport only the decommissioning radioactive waste transport container, which is the same size as the IP-2 transport container, in the cargo hold.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No.20211510100040).

Keywords: CHEONG JEONG NURI, Transport Container, Cargo Arrangement

Scale-up Study for Volume Reduction Process of Radioactive Concrete Waste From Decommissioning Projects

Maengkyo Oh^{1,2}, Keunyoung Lee^{2,*}, Min-Ku Jeon², Richard Ian Foster^{2,3}, and Chang-Ha Lee¹

¹Yonsei University

²Korea Atomic Energy Research Institute (KAERI)

³Seoul National University (SNU)

*lky@kaer.re.kr

Decommissioning waste is generated at all stages during the decommissioning of nuclear facilities, and various types of radioactive waste are generated in large quantities within a short period. Concrete is a major building material for nuclear facilities. It is mixed with aggregate, sand, and cement with water by the relevant mixing ratio and dried for a certain period. Currently, the proposed treatment method for volume reduction of radioactive concrete waste was involved thermomechanical and chemical treatment sequentially. The aggregate as non-radioactive materials is separated from cement components as contaminated sources of radionuclides. However, to commercialize the process established in the laboratory, it is necessary to evaluate the scale-up potential by using the unit equipment. In this study, bench-scale testing was performed to evaluate the scale-up properties of the thermomechanical and chemical treatment process, which consisted of three stages (1: Thermomechanical treatment, 2: Chemical treatment, 3: Wastewater treatment). In the first stage, lab, bench, and pilot scale thermomechanical tests were performed to evaluate the treated coarse aggregate and fines. In the second stage, the fine particles generated by the thermomechanical treatment process, were chemically treated using dissolution equipment, after then the removal efficiency and residual of cement in the small aggregate was compared with laboratory results. The final stage, the secondary wastewater containing contaminant nuclides was treated, and the contaminant nuclides could be removed by chemical precipitation method in the scale-up reactors. Furthermore, an additional study was required on the solid-liquid separation, which connected each part of the equipment. It was conducted to optimize the separation method for the characteristics of the particles to be separated and the purpose of separation. Therefore, it is expected that the basic engineering data for commercialization was collected by this study.

Acknowledgements

This work was supported by the National Research Foundation of Korea grant (No. RS-2022-00155421) funded by the Ministry of Science and ICT.

Keywords: Radioactive Concrete, Volume Reduction, Bench-scale, Thermomechanical Process, Chemical Treatment, Phase Separation

Technology Development for Advanced Volume Reduction of Radioactive Concrete Waste From NPP Decommissioning

Keunyoung Lee^{1,*}, Maengkyo Oh^{1,2}, Richard I. Foster^{1,3}, Kwang-Wook Kim¹, Sungjune Sohn¹, Min Ku Jeon¹,
and Bum-Kyoung Seo¹

¹Korea Atomic Energy Research Institute (KAERI)

²Yonsei University

³Seoul National University (SNU)

*lky@kaeri.re.kr

Radioactive waste generated in large quantities from NPP decommissioning has various physicochemical and radiological characteristics, and therefore treatment technologies suitable for those characteristics should be developed. Radioactively contaminated concrete waste is one of major decommissioning wastes. The disposal cost of radioactive concrete waste is considerable portion for the total budget of NPP decommissioning. In this study, we developed an integrated technology with thermomechanical and chemical methods for volume reduction of concrete waste and stabilization of secondary waste. The unit devices for the treatment process were also studied at bench-scale tests.

The volume of radioactive concrete waste was effectively reduced by separating clean aggregate from the concrete. The separated aggregate satisfied the clearance criteria in the test using radionuclides. The treatment of secondary waste from the chemical separation step was optimally designed, and the stabilization method was found for the waste form to meet the final disposal criteria in the repository site. The final volume reduction rates of 56.4~75.4% were possible according to the application scenario of our processes under simulated conditions. The commercial-scale system designs for the thermomechanical and chemical processes were completed. Also, it was found that the disposal cost for the contaminated concrete waste at domestic NPP could be reduced by more than 20 billion won per each unit. Therefore, it is expected that the application of this technology will improve the utilization of the radioactive waste disposal space and significantly reduce the waste disposal cost.

Acknowledgements

This work was supported by the National Research Foundation of Korea grant (No. RS-2022-00155257 & RS-2022-00155421) funded by the Ministry of Science and ICT.

Keywords: Decommissioning, Radioactive Concrete Waste, Volume Reduction, Stabilization, Disposal

Demonstration Test of Full System Decontamination and Its Operational Review

Sung Hyun Lee, Jung-Hyun Lee, Ju-Hyeon Park, Sung-Sik Yoon, Jeong-Yeop Cha, Seung-Ki Lee,

Sung-Su Kim, Chung-kyu Lee, and Ki-Chul Kim*

KEPCO Korea Plant Service & Engineering (KEPCO KPS)

*kichkim@kps.co.kr

In this research, KPS manufactured Full System Decontamination (FSD) equipment, which is consisted of Oxidizing Agent Manufacturing System (OAMS), Chemical Injection System (CIS), RadWaste Treatment System (RWTS), Chemical Waste Decomposition & Treatment System (CWDS) and conducted demonstration test to prepare Decontamination and Decommissioning (D&D) project of Kori nuclear power plant in Korea.

Each equipment of FSD was modularized due to the limited size of equipment hatch of Kori nuclear power plant. To simulate the expected circumstances in nuclear power plant such as usage of heater or position of each equipment, additional equipment was used. The chemical concentration and flow rate of process water for FSD were used as mentioned in the previous study by KHNP CRI.

FSD was conducted for three cycles and each cycle was consisted of oxidation, reduction, chemical decomposition and purification. Oxidation and reduction process were conducted at 90°C. Chemical decomposition and purification process were conducted at 40°C due to the damage of UV lamp and IX by the heat. Total volume of process water for FSD demonstration test was 2.5 m³.

KPS conducted decontamination performance review by calculating thickness reduction and weight loss of installed specimen. Operational review was conducted as if FSD test was conducted in the field based on the result of demonstration test. One of the most prioritized features is the workers' safety. Also, the appropriate position of equipment needs to be considered to meet the required specification of component.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2019 1510301310).

Keywords: Full System Decontamination, Operational Review, Modularized Equipment, Decommissioning

A Study on the Sintering Stabilization Method of Cs-contaminated Spent Adsorbent

Keunyoung Lee^{1,*}, Maengkyo Oh^{1,2}, Jaewoong Hwang^{1,3}, and Min Ku Jeon¹

¹Korea Atomic Energy Research Institute (KAERI)

²Yonsei University

³Chungnam National University (CNU)

*lky@kaeri.re.kr

This study evaluated the synthesis of optimal materials for high efficiency adsorption and removal characteristics of Cs-137 for radioactive contaminated water, and considered thermal treatment methods to stabilize the spent adsorbent generated after treatment. We synthesized a composite adsorbent with a combination of impregnating metal ferrocyanide that improves the selectivity of Cs adsorption with zeolite capable of removing Cs as a support. The Cs removal efficiency of the composite adsorbent was evaluated, and the stability change of Cs according to the high-temperature sintering was evaluated as a stabilization method of the spent adsorbent.

The metal ferrocyanide content of the adsorbent was in the range of 11.8~36.0%. The adsorption experiments were performed using a simulated liquid waste to have a total Cs concentration of 1 mg/L while containing a trace amount of Cs-137, and then gamma radioactivity was analyzed. In order to evaluate the stabilization of the spent adsorbent, heat treatment was performed in the range of 500~1,100°C, and the volatilization rate of Cs during heat treatment and the leaching rate of Cs after heat treatment were compared.

In the adsorption experiment, the Cs removal efficiency was higher than 99%, regardless of the amount of metal ferrocyanide in the composite adsorbent. In the sintering experiment on the spent adsorbent, it was confirmed that there was no volatilization of Cs up to 850°C, and then the volatilization rate increased as the heating temperature increased. On the other hand, the leaching rate of Cs in the sintered adsorbent tends to significantly decrease as the heating temperature increases, so that Cs can be stabilized in the sintered body. In addition, as the content of metal ferrocyanide increases, the volatilization rate of Cs rapidly increases, indicating that the unstable metal ferrocyanide in the adsorbent may adversely affect the removal of Cs as well as the thermal treatment stability.

Acknowledgements

This work was supported by the National Research Foundation of Korea grant (No. RS-2022-00155257 & RS-2022-00155421) funded by the Ministry of Science and ICT.

Keywords: Decommissioning, Radioactive Cesium, Adsorption, Thermal Treatment, Stabilization

A Study of Effect of Trace Materials in Activation Analysis

Mi-Hyun Lee*, Young Hwan Hwang, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*mihyun.lee@khnp.co.kr

The segmentation of activated components including reactor vessel and reactor vessel internals requires many information. The primary information is material composition, trace materials in the composition, neutron flux during operation, etc. According to the EPRI report the primary basis of activity in a decommissioning source term is the activated metals from the reactor vessel and vessel internal components. The report indicates that over 95% of the radioactivity from decommissioning, except from spent nuclear fuel, consists of activated metals. These are from the reactor vessel, reactor internal structures and expendable components which are constructed primarily of various grades of stainless steel. Stainless steel contains appreciable levels of impurity cobalt. The common primary radionuclides of concern for the disposal environment from activated metals identified in US and international studies include C-14, Cl-36, Ni-59, Co-60, Ni-63, etc. The most common types of stainless steels used in reactor vessel construction and internal components include the Type 304(L), Type 316(L) and various grades of Inconel. The components of stainless steel are mainly Ni, Cr, Mo, Nb, etc., and when these elements are activated, they produce nuclides such as Nb-94, Tc-99, Sr-90, etc.

In this study, the current status of activation analysis is reviewed to understand the effects of many variables. Also, the effect of trace materials is reviewed, including transformation of radioactive nuclides.

Keywords: Activation Analysis, Trace Materials, Stainless Steel, Segmentation

A Study on Waste Management of ZION Nuclear Power Plant Decommissioning

Young Hwan Hwang*, Mi-Hyun Lee, and Seok-Ju Hwang

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*yhhwang7@khnp.co.kr

The decommissioning of nuclear power plant (NPP) consists of various activities, such as system decontamination, take out of activated components, segmentation of the activated components, site remediation, etc. During various activities, the generation of radioactive wastes and radiation exposure to workers is expected. The systematic waste management during the activities is important to implement the decommissioning. The inefficient waste management usually brings significant delay in the decommissioning process and results in an increase of decommissioning cost. The radiation exposure management is also an important issue. It is generally accepted that the hot spot, generated from operation and decommissioning of NPP, is observed in many places within the containment building. Although the health physicists measure the radiation in various points, the unintended hot spots are sometimes generated and observed. The effective radiation exposure management also requires the control of personnel and space during various activities.

In this study, the radiation exposure and waste management experiences of Zion NPP are reviewed. The primary nuclides and radiation exposure during various activities are systematically studied to achieve the main objectives of this paper.

Keywords: ZION NPP, Waste Management, Radiation Exposure, Decommissioning

Study on Creating and Optimizing a Task Sequence for Robotic Underwater Laser Cutting Using a Digital Manufacturing System

Jaehyun Ha*, Dongjun Hyun, Sungmoon Joo, Ikjune Kim, and Jonghwan Lee

Korea Atomic Energy Research Institute (KAERI)

*jhha78@kaeri.re.kr

This study presents an example of creating and optimizing a task sequence required in an automated remote dismantling system using a digital manufacturing system. An automated remote dismantling system using a robotic arm has recently been widely studied to improve the efficiency and safety of the dismantling operations. The task sequence must be verified in advance through discrete event-based process simulation in a digital manufacturing system to avoid problems in actual remote cutting operations as the main input of the automated remote dismantling system.

The laser cutting method can precisely cut complicated target structures such as reactor internals with versatility, but a robot and a pre-prepared program are required to deploy sophisticated motion of the laser cutting head on the target structure. For safe and efficient dismantling operations, the robot's program must be verified in advance in a virtual environment that can represent the actual dismantling site.

This study presents creating and optimizing the task sequence of a robotic underwater laser cutting as part of the project of developing an automated remote dismantling system. A task sequence is created to implement the desired cutting path for the target structure using the automated remote dismantling system in the virtual environment. The task sequence is optimized for the posture of the laser cutting head and the robot to avoid collisions during the operation through discrete event-based process simulation since the target structure is complicated and the volume occupied by the laser cutting head and the robot arm is considerably large.

The task sequence verified in the digital manufacturing system is demonstrated by experiments cutting the target structure along the desired cutting path without any problems. The various simulation cases presented in this study are expected to contribute not only to the development of the automated remote dismantling system, but also to the establishment of a safe and efficient dismantling process in the nuclear facility decommissioning.

Keywords: Remote Dismantling System, Digital Manufacturing System, Laser Cutting, Robot, Underwater

Performance Tests on the Plasma Arc Melter-Molten Salt Oxidizer Combination System – 2. Waste Form Integrity of Melting Solidification Products

In Sun Jang^{1,*}, Na Young Kim¹, Jae Hyuk Choi¹, Hee Chul Yang², and Hee Chul Eun²

¹Dawonsys

²Korea Atomic Energy Research Institute (KAERI)

*isjang@dawonsys.com

Present study investigated the waste form integrity of melted products generated from PAM-MSO system, which is proposed and developed to compensate the drawbacks of each system. The disposal suitability of the melting solidification products generated from the plasma arc melting treatment of pulverized cement debris spiked by Pb, Cd and Cs, as indicators of typical hazardous metals and radionuclides existed in the low-level mixed waste in the KHNPPs. The final waste form obtained by the test was evaluated for suitability for disposal. The compressive strength was 261.10 MPa, showing much higher values when compared to other waste form products. The compressive strength of both the sample after irradiation with 10^7 Gy radiation and that after long-term submersion test (90 days) satisfied the disposal criteria. As a result of the leaching test conducted according to the ANS 16.1 test method, it was confirmed that the leaching index satisfies the disposal criteria.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 3210100050, Treatment Technology Development and Disposal Suitability Evaluation for Radioactive Mixed Wastes).

Keywords: Plasma Arc Melter, Molten Salt Oxidation System, Mixed Waste, Waste Form, Disposal Suitability

Performance Tests on the Plasma Arc Melter-Molten Salt Oxidizer Combination System – 1. Destruction of Organics and Retention of Metals/Radionuclides

In Sun Jang^{1,*}, Na Young Kim¹, Jae Hyuk Choi¹, Hee Chul Yang², Hee Chul Eun², and Hyung Ju Kim²

¹Dawonsys

²Korea Atomic Energy Research Institute (KAERI)

*isjang@dawonsys.com

Plasma Arc Melter (MSO) system has been developed for the treatment and the stabilization of various kinds of hazardous and radioactive waste into the readily disposable solidification products. Molten salt oxidation system has been developed for the treatment of halogen- and sulfur-bearing hazardous and radioactive waste without emissions of PCDD/Fs and acid gases. However, PAM system has showed some difficulty in the off-gas treatment system due to the volatilization of radionuclides and toxic metals at extremely high-temperature plasma arc melter and the emissions of acid gases. MSO system has also showed the difficulty in the treatment of spent molten salt into the disposable waste form.

Present study discussed the results of organics destruction performance tests for the PAM-MSO combination system, which is proposed and developed to compensate the drawbacks of each system. The worst-case condition tests for the organics destruction were conducted at lowest temperatures and the worst-case condition tests for the retention of metals and radionuclides were conducted at highest temperatures under the range of normal operating condition.

For the worst-case organic destruction test, C₆H₅Cl was selected as a POHCs (Principal Organic Hazardous Constituents) because of its high incinerability ranking and the property of generation of chlorine gases and PCDD/Fs when incompletely destroyed. Simulated concrete waste spiked with 1 L of C₆H₅Cl was treated and the emissions of 17 kinds of PCDD/Fs and other hazardous gases such as CO, THC, NO_x, SO₂ and HCl/Cl₂ were measured. For the worst-case condition tests for the retention of metals and radionuclides, Pb and Cs were selected because of its high volatility characteristics.

The emissions of PCDD/Fs was extremely lowered than the emission limit and those of other hazardous constituents were below their emission limit. The results of performance tests on the organics destruction suggested that tested PAM-MSO combination system could readily treat PCBs-bearing spent insulation liquid, spent ion-exchange resins used for the treatment of spent decontamination liquid in the decommissioning process and the concreted debris bearing hazardous organic coating materials. The decontamination factor of Cs and Co were 1.4×10^5 , 1.4×10^5 , respectively. The emission of Pb was 0.562 ppm. These results suggested that tested PAM-MSO system treated low-level radioactive and Pb-bearing mixed waste.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 3210100050, Treatment Technology Development and Disposal Suitability Evaluation for Radioactive Mixed Wastes).

Keywords: Plasma Arc Melter, Molten Salt Oxidation System, Mixed Waste, Decommissioning Waste, Vitrification

Study on the Decomposition for Radioactive Wastes Containing a Hazardous Material (Asbestos) in a High-Temperature Melt Oxidation System

Na-Young Kim^{1,*}, In-Sun Jang¹, Jae-Hyuk Choi¹, Hee-Chul Yang², and Hee-Chul Eun²

¹Dawonsys

²Korea Atomic Energy Research Institute (KAERI)

*nykim@dawonsys.com

Radioactive mixed waste (RMW) is containing radioactive materials and hazardous materials. Radioactive wastes containing asbestos are include in RMW. These wastes thus must be treated considering both radioactive and hazardous aspects.

In this study, a high temperature melt oxidation system consisting of an electric arc furnace and a molten salt oxidation furnace has been developed for the treatment of of radioactive waste containing asbestos. A surrogate waste of the radioactive waste containing asbestos (content of asbestos: 13wt%) was treated in this system. It was melted and fabricated into a glass waste form in the system. Asbestos was not detected in this glass waste form. This means that the asbestos was converted to a glass component in the glass waste form. The waste form was homogeneous glass, and it had a high value of compressive strength (475.13 MPa). It was also confirmed through a leaching test (ANS 16.1) that the waste form had a high chemical durability (Leaching Index > 6).

Based on these results, it is considered that the high temperature melt oxidation system will be utilized for the treatment of a significant amount of radioactive waste containing asbestos generated from decommissioning a nuclear power plant.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2020 3210100050, Treatment Technology Development and Disposal Suitability Evaluation for Radioactive Mixed Wastes).

Keywords: Hazardous Material, Asbestos, Heat Treatment, Molten Salt Oxidation

Separation of Hydrogen Isotopes by Water Electrolysis

Chan Woo Park* and Ilgook Kim

Korea Atomic Energy Research Institute (KAERI)

*chanwoo@kaeri.re.kr

Water electrolysis is an efficient method to enrich heavy hydrogen isotopes (tritium and deuterium) in the aqueous phase. Although an alkaline water electrolyzer has been commercialized for mass production of hydrogen, such a method requires additional purification steps to remove electrolytes from the final concentrates. On the other hand, proton exchange membrane water electrolysis (PEMWE) does not require additional electrolyte treatment steps, and PEMWE is operated at higher current density compared to the alkaline water electrolysis. In this study, we investigated deuterium and tritium separation from light water by PEMWE. Separation behaviors at the anode and cathode were analyzed, and H/D and H/T separation factors were compared.

Keywords: Tritium, Deuterium, Detritiation, Water Electrolysis

Conflicts Over the Clearance Waste Recycling From the Decommissioning of Nuclear Power Plants

Ye-Jeon Lim*, Dong-Jin Kim, So-On Park, and Byung-Sik Lee

Dankook University

*akwlakrrhc@naver.com

In this study, a manual that can be applied to conflict management of clearance waste recycling by stakeholders was researched to recycle clearance waste that is most frequently generated when decommissioning nuclear power plants. In order to develop a manual that can be applied to conflict management, the content of the conflict should be derived first. In order to derive conflict, it is necessary to organize major issues in recycling clearance waste in consideration of domestic nuclear energy and social environment. In order to organize major issues in consideration of the domestic environment, a literature survey and a domestic current situation investigation were conducted. At this time, the subject of the major issue was selected based on the Level 1 influencing factors of the previous study. As a result of the investigation, it was confirmed that there were many major issues due to lack of reliability/understanding in nuclear energy/radiation. Through this Conflicts caused by recycling clearance waste were derived based on the organized issues.

As a result of deriving conflicts, eight conflicts were derived below.

- 1) Reduced business availability due to lack of understanding/reliability
- 2) Lack of reliability in the selection and technology of nuclide analysis technology
- 3) Additional time and equipment required due to establishment of clearance waste regulatory requirements
- 4) Low economic benefits due to reduction in the effect of substituting raw materials
- 5) Political interference due to worsening public opinion
- 6) Rejection of final products due to recycling due to distrust of radiation
- 7) Public acceptance along the transport route from the source to the recycling plant
- 8) Business promotion deteriorated due to changes in energy policy

As a result of the derived conflict analysis, the most conflicts related to lack of reliability/understanding in nuclear energy/radiation were derived. Accordingly, in future research, it is necessary to prepare a specific plan to enhance the understanding of stakeholders about self-disposal waste recycling.

Considering that research that can solve the conflicts that will be faced when the domestic/foreign clearance waste recycling industry is activated is not activated, this study is meaningful in that it derived the conflicts that will be faced when recycling clearance waste. Also, it is expected that the conflicts derived from this study will be used meaningfully in the establishment of the clearance waste recycling management manual.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP), granted financial resource from the Ministry of Trade, Industry and Energy (No. 20203210 100430), Republic of Korea.

Keywords: Decommissioning, Clearance Waste, Conflict Management, Conflict Derivation

A Revised Method for Estimating Initial Provision of Nuclear Power Plant (NPP) Decommissioning Cost

Seung-Hyuk Jung* and Hyun-Woo Song

KONES Corp.

*ideabizman@kones21.com

Under Article 17 of the Radioactive Waste Management Act and Article 12 of the Enforcement Decree of the Radioactive Waste Management Act, KHNP shall reserve the cost for the decommissioning of NPPs as provisions. To preserve the value, an additional amount considering the discount rate is to be added annually. The initial provision is decided by estimating the decommissioning cost of NPP at the time of commercial operation, calculating the future cost by applying the inflation rate to the expected start date of decommissioning, and then discounting it at a discount rate to the present value. According to the current notice, the period for applying inflation and discount rate is defined as the period of 5 years added to the design life of NPP, which is presumed to be due to the assumption that all decommissioning costs are incurred at once 5 years after the permanent shutdown of the power plant. However, assuming that the actual decommissioning period of a domestic nuclear power plant is generally planned for 15 years, it can be expected that most of the decommissioning activities will begin after the decommissioning preparation and transition period, or 5 years after permanent shutdown of the plant. Considering this, it can be said that the current period (5 years + design life) for applying inflation and discount rate is set a little conservatively.

In this paper, the initial provision is calculated by appropriately distributing the decommissioning costs of overseas NPPs categorized by International Structure for Decommissioning Costing (ISDC) during the planned decommissioning period of domestic NPPs, and then adding up the decommissioning cost each year by separately applying the inflation and discount period, which was compared with the results calculated using the current method. Through this, it was confirmed that the revised method had the effect of reducing the initial provision by 2.2% to 5.7% compared to the current method depending on the gap between inflation rate and discount rate, which can be converted to about 8 years of inflation and discount period used in the current method. It is expected that this paper will be used in the future as a basic reference for developing a more accurate method for calculating the initial provision of decommissioning cost.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted fund by MOTIE, Republic of Korea (No. 20211510100040).

Keywords: Decommissioning Cost, Annual Cost, Initial Provision, Decommissioning Period, Inflation Rate, Discount Rate

A Preliminary Modeling to Characterize Major Factors Affecting Neutron Activation of Medical Cyclotrons Using MCNP, FISPACT-II and FLUKA

Se Geun Lee and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Medical cyclotrons have been used for dedicated medical or commercial applications such as positron emission tomography (PET) for the past tens of years. These cyclotron facilities have produced positron-emitting radionuclides (i.e. ^{11}C , ^{13}N , ^{15}O , ^{18}F , etc.). Among them, ^{18}F , produced by $^{18}\text{O}(\text{p},\text{n})^{18}\text{F}$ reaction is the most widely used which has longer half-life (around 110 m) and lower energy of emitted positrons (around 0.63 MeV). Secondary neutrons produced during $^{18}\text{O}(\text{p},\text{n})^{18}\text{F}$ reaction could cause neutron activation of structures, systems, and components of cyclotron facilities.

Therefore, International Atomic Energy Agency (IAEA) had addressed that during the operation of cyclotrons, concrete walls become radioactive over time and this radioactivity needs to be characterized for planning of the facility decommissioning. Moreover, several prior studies had estimated the neutron activation and levels of radioactivity of concrete wall of cyclotron facilities. Although those studies assessed the neutron activation of actual cyclotron facilities, however, the purpose of assessment was only for decommissioning each individual facility. Also, the assumptions, conditions or insights of conclusion may be limited to each individual case. For these reasons, this study focused on analysis of effects of major factors (e.g. concrete type, impurity contents of structural materials, etc.) about neutron activation of cyclotron facilities.

In this study, the well-known methodology of neutron activation estimation was established and neutron activation products of concrete wall of cyclotron vault was calculated. Also, sensitivity analyses were conducted to figure out the effects of major factors of neutron activation and production of radioactive wastes during decommissioning of the facility. The methodology and results were validated by two steps: comparing with prior studies and comparing with another computer code.

Concrete type did not affect that the decision of level of radioactivity waste criteria. Because of relatively longer half-lives, impurity contents of structural materials especially Co and Eu were turned out one of the most important factors for planning the facility decommissioning. It is hard to simply figure out the radioactivity levels of cyclotron facilities, however, rough predictions of minimum period for decay-in-storage as radioactive waste management can be possible with using information of thermal neutron spectra and major impurity nuclides (e.g. ^{59}Co , ^{151}Eu and ^{153}Eu) for minimization of radioactive waste production and relief of charge of radioactive waste management.

Acknowledgements

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of the Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 2021040101002B).

Keywords: Neutron Activation, Radioactive Waste Management, Monte Carlo, MCNP, FISPACT, FLUKA

6분과

방사선환경 및 안전(Oral)

Review on the Evaluation Methodology of Radioactive Source Term for Normal Operation of NuScale SMR

Ji-Young Song, Gyeong-Mi Kim, Su-Ji Yoon, Ho-Jin Lee*, and Kwan-Hee Lee

Korea Institute of Nuclear Safety (KINS)

*k708lhj@kins.re.kr

Radioactive source terms are important factor in design, licensing and operation of SMR (Small Modular Reactor). In this study, regulatory requirements and evaluation methodology for normal operation on NuScale SMR, which received standard design certification approval on September 11, 2020 from US NRC, are reviewed.

The radioactive waste management system of nuclear power reactor should be designed to limit radionuclide concentration in effluents and keep radioactive effluents at restricted area boundary ALARA according to 10 CFR 20 and 10 CFR 50 Appendix I. Also, in general, the coolant source term to calculate the off-site radiological consequences for normal operation of SMR should be determined by using models and parameters that are consistent with regulatory guide 1.112, NUREG-0017 and the guidance provided in ANSI/ANS-18.1-1999, and the result should be corrected by reflecting the design characteristics of SMR.

The coolant source term of NuScale, unlike the case of large NPPs, cannot rely solely on empirical source term data, because the NuScale source term is based on first principle physics, operational experience from recent industry, and lessons learned from large PWR operation.

Fission products in reactor coolant are conservatively calculated using first principle physics in SCALE Code assuming 60 GWD/MTU. The release of fission products from fuel to primary coolant based on industry operational experience is determined as fuel failure fraction of 0.0066% for normal operation source term and 0.066% for design basis source term while coolant source term of large NPP is calculated by using ANSI/ANS-18.1 for normal operation and fuel failure fraction of 1% for design basis source term.

Water activation products in reactor coolant are calculated from first principles physics and corrosion activation products are calculated by utilizing current large PWR operating data (ANSI/ANS 18.1-1999) and adjusted to NuScale plant parameters. Also, because ANSI/ANS 18.1-1999 is not based on first principle physics models for CRUD generation, buildup, transport, plate-out, or solubility, NuScale has incorporated lessons learned by using ERPI's primary water chemistry and steam generator guidelines to ensure source term is conservative and design of materials used cobalt reduction philosophy to help ensure the coolant source term are conservative.

Based on the coolant source term calculated according to the above-described method, the annual releases of radioactive materials in gaseous and liquid effluents from NuScale reactor are evaluated.

Currently, Small Modular Reactors such as ARA, SMART 100 are under review for licensing in Korea. This study will be helpful to understand how the reactor coolant system source terms are defined and evaluated for SMR.

Keywords: Coolant Source Term, Normal Operation Source Term, Design Basis Source Term, SMR, ANSI/ANS-18.1

Radiation Dose Assessment of Thorium-containing Gas Mantle for Use and Disposal Scenarios

Jimin Shin¹, Beomseok Ko¹, Hee Seo^{1,*}, Meeryeong Kim², Minkyung Kim², and Sangmin Lee²

¹JeonBuk National University (JBNU)

²Korea Institute of Nuclear Safety (KINS)

*hseo@jbnu.ac.kr

The IAEA recommended considerations for exemption regulations of consumer products containing greater amounts of radioactive isotopes than the amounts specified for generic exemption. One of the major considerations is the expected exposure dose should be less than 10 $\mu\text{Sv/y}$ and 1 mSv/y for general cases and low probability cases, respectively, in all predictable scenarios. Under this recommendation, many countries evaluated the radiation dose for exposure scenarios of various products in consideration of the national circumstances and, then, established their own specific exemption regulation. In Republic of Korea, the “Regulation on substances excluded from radioactive isotopes” was legislated to specify consumer products excluded from regulation. However, as the usage status and product specifications has changed over time, it is necessary to periodically verify the validity of the regulation criteria in the view of exemption justification. In this study, we developed the use and disposal scenarios in consideration of the domestic use of thorium-containing gas mantle and evaluated radiation dose of each scenario accordingly. The gas mantles are used as a wick for gas lanterns and the maximum activity of natural thorium contained among the currently available gas mantles is 12.5 kBq. Radioactive isotopes in the decay chain of natural thorium can be divided into three groups according to their physical characteristics, and exposure routes suitable for each group were considered in dose calculation. Currently, most gas mantles are installed in camping lanterns. Therefore, we developed use scenarios related to camping. The average number of camping trips and time spent at the campground were set by the data from Korea Tourism Organization. Tent sizes and vehicle specifications were determined by referring to surveys and products in Korea. The used gas mantle is disposed of in a garbage bag for general waste and transported to landfill or incinerator. We determined the amount of gas mantle discarded in landfill and incinerator by the data from Korea Environment Corporation. The exposure time and amount handled by an individual were determined by considering the number of waste collection vehicles, landfills, and incinerators. Although we assumed the maximum activity of the gas mantle for conservative evaluation, the calculated radiation doses for the use and disposal scenarios were below the general requirement (i.e., 10 $\mu\text{Sv/y}$) in all scenarios.

Acknowledgements

This study was supported by the Korea Institute of Nuclear Safety.

Keywords: Consumer Product, Radioactive Isotope, Gas Mantle, Exposure Scenario, Dose Assessment

Verification of Monte Carlo Simulation Model on Seafood Monitoring System for Radioactive Material Screening

Hyosang Choi and Hee Seo*

JeonBuk National University (JBNU)

*hseo@jbnu.ac.kr

Considering the Fukushima nuclear accident and the marine discharge plan of contaminated (or treated) water, it is necessary a seafood monitoring system for radioactive material screening. Currently, radioactivity tests in seafood are conducting in Korea. Although current method using a HPGe detector can provide very low uncertainty in determining radioactivity, there is a limitation in that rapid inspection cannot be performed because of a time-consuming pretreatment process as well as long measurement time (typically 10,000 s). To overcome this limitation, we are developing an in-situ inspection device, a kind of screening system, which can monitor the radioactivity in seafood in a near real-time basis. In this study, the actual seafood with a check source was measured to verify the reliability of the Monte Carlo simulation model. The detector used in the experiment was a 5-cm-thick polyvinyl toluene (PVT) plastic scintillator with a 0.5-cm-thick lead shield for background reduction. A Cs-137 check source was placed within seafood. The seafood used in the experiment was fishcake, raw oyster, and dried laver, which is the representative seafood of fish, shellfish, and seaweed. These three seafood products of the same size and shape as the manufacturing process were used to predict the performance realistically. We compared the energy spectrum of the Cs-137 check source obtained from measurements and simulations. The region of interest (ROI) around the Compton edge was set to reduce the influence of the background and electronic noise. The results showed that the measured and simulated spectrum were in good agreement.

Acknowledgements

This research was a part of the project titled ‘Development of Smart Processing Technology for Sea Foods’, funded by the Ministry of Oceans and Fisheries, Korea.

Keywords: Seafood, Monitoring, Radioactive Material, Monte Carlo

Radiation Safety Evaluation for Radioactive Waste Processing in Hot Cell Facility

Kyung Hun Shin and Hee Reyoung Kim*

Ulsan National Institute of Science and Technology (UNIST)

*kimhr@unist.ac.kr

In Korea, Kori Unit 1, a commercial pressurized water reactor (PWR), was permanently shut down in June 2017, and an immediate decommissioning strategy is underway. Therefore, it is essential to understand the characteristics of radioactive waste during the decommissioning process of nuclear power plants (NPP). Because radioactive waste must be handled with care, radioactive waste is treated in a hot cell facility. Hot cell facility handles radioactive waste, and worker safety is essential. In this study, it was dealt with whether or not the radiation safety regulations were satisfied when processing the core beltline metal of the dismantling waste treated at the post irradiation examination facility (PIEF) of the hot cell facility. Core beltline metal used for the pressure vessel in the reactor is carbon steel, and it is continuously irradiated by neutrons during the operation of the NPP. A radiological safety estimation of the behavior of radioactive aerosols during the cutting process within the PIEF was carried out to ensure the safety of the environment and workers. When processing the core beltline metal in PIEF, dominant six nuclides (^{60}Co , ^{63}Ni , ^{55}Fe , ^3H , ^{59}Ni , ^{14}C) of aerosol are generated. Accordingly each cutting device, amount of aerosol and value of dose is different. Using a 99.97% efficiency HEPA filter, the emission concentration of the dominant nuclides (^{60}Co , ^{63}Ni , ^{55}Fe , ^3H , ^{59}Ni , ^{14}C) in the air source term was satisfied with the emission control standard of Nuclear Safety Commission No. 2016-16. It was confirmed that the radioactivity concentration in the airborne source term inside the PIEF is in equilibrium state, when ventilation is considered. Also, the mass of aerosol and the concentration of airborne source term differed according to the thickness of the saw blade of the cutting tool, and the exposure dose of the worker was different through Monte Carlo N-Particle (MCNP). At that time, ^{60}Co accounted for 95.4% of the exposure dose, showing that ^{60}Co had the highest impact on workers, followed by ^{55}Fe with 2.7%. The worker's dose limit is satisfied in accordance with Article 2 of the Nuclear Safety Act and the dose limit of radiation-controlled area is found to be satisfied in accordance with Article 3 of the rules on technical standards for radiation safety management at this time.

Acknowledgements

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (No.20201520300009, Safety evaluation and design technology development of core demonstration facilities for nuclear power plant decommissioning). This work was supported by the Korea Institute of Energy Technology Evaluation and Planning and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (grant no. 20214000 000410).

Keywords: Hot Cell Facility, Ventilation, Airborne Source Term, Nuclear Safety Act

LSC Quenching Correction of H-3 Leached Sample of Paper Packaging Material

Changhyun Ryu^{1,2}, Sangyong Park¹, Junghwan Hong¹, Jusngsuk Park¹, and Kwangpyo Kim^{2,*}

¹KEPCO Nuclear Fuel (KEPCO NF)

²Kyung Hee University

*kpkim@khu.ac.kr

In liquid scintillation counting, sample radioactivity is analyzed by measuring photons emitted from counting vials. Quenching effect lowers photon intensity from samples, which leads to lower counting efficiency. So an appropriate quenching correction according to characteristics of samples is important. In this study, the quenching correction for H-3 analysis was conducted according to the characteristics of paper packaging material leached samples. The leached samples are made from H-3 leaching method which is in the process of development for H-3 contamination screening.

There are several ways of quenching correction such as internal standard (IS) method, quench correction curve and triple-to-double coincidence ratio (TDCR) method, etc. For quench correction curve, quenched standard set, which has the same matrix as experimental samples, is needed to be prepared. Each leached sample, however, has different matrix and color depending on condition of leaching experiment, which means that it is not capable of preparing standard set having same matrix with the samples. In this study, the counting samples are used for plotting quench correction curve instead of quenched standard set. Spectral quench parameter of the external standard [SQP(E)] is used as quench indicating parameter (QIP). TDCR and counting efficiencies determined by IS method are used as counting efficiencies. The quench curve of TDCR versus SQP(E) has $R^2 = 0.55$ and the curve of efficiency from IS method versus SQP(E) has $R^2 = 0.99$. TDCR is known for approximate counting efficiency, however, TDCR as counting efficiency needs careful use for H-3 analysis of leached samples. The curve used efficiency from IS method is suitable for H-3 analysis of leached samples.

In this study, the quench correction curve is prepared for H-3 analysis of leached samples of paper packaging material. SQP(E), TDCR and efficiency from IS method was used as parameters to plot the quench correction curve, and, the efficiency from IS method is suitable for H-3 analysis of the leached samples. The result of this study can be used for H-3 analysis of leached samples of paper packaging material.

Keywords: Liquid Scintillation Counting, Quenching Correction, TDCR, Internal Standard Method

Shielding Thickness of the Fuel Handling Hot Cell at the Spent Fuel Re-packaging Facility

Joon Gi Ahn*

KEPCO Engineering & Construction Company (KEPCO E&C)

*jgahn@kepc-co-enc.com

For deep geological repository of the spent nuclear fuel, the fuel assemblies loaded in the storage cask are transferred to the disposal cask and the operation is performed in the fuel handling hot cell at the fuel re-packaging facility. As the fuel handling hot cell shielding is accomplished by the concrete wall and the viewing glass window, the required shielding thickness was evaluated for both materials. The ordinary concrete is applied to hot cell wall and two kinds of glasses, i.e., single layer of lead glass and double layer of lead glass and borosilicate glass, are considered for the viewing glass window. A bare spent PWR fuel assembly exposed to the environment in the hot cell was considered as the neutron and gamma radiation sources. The neutron and gamma transport calculations were performed using the MAVRIC program of the SCALE code system for the dose rate evaluation. The dose limit of 10 $\mu\text{Sv/h}$ is applied as the target dose to establish the required shielding thickness.

The concrete wall of 94 cm thickness reduces the total dose rate to 6.9 $\mu\text{Sv/h}$, which is the sum of neutron dose and gamma dose. Penetrating the concrete wall, both of the neutron dose and the gamma dose decrease constantly with shield thickness and the gamma dose is always dominant through whole penetrating distance. Single layer lead glass of 74 cm thickness reduces total dose rate to 6.2 $\mu\text{Sv/h}$. Applying double layer shield glass combined of lead glass and borosilicate glass, the total dose rate reduces to 3.6 $\mu\text{Sv/h}$ at same shield thickness of 74 cm. Through the shield glass, gamma dose decreases rapidly and neutron dose decreases slowly compared with those for concrete wall. In result, neutron dose becomes dominant on the window glass shielding. The more efficient dose reduction of double layer glass is achieved by the borosilicate glass's superior neutron shielding power. Thus, the use of double layer glass of lead glass and borosilicate glass is recommended for the viewing glass of the fuel handling hot cell. Finally, it is concluded that about 1 m thick concrete wall and 75 cm thick viewing glass window are sufficient for the radiation shielding of the hot cell at the spent fuel re-packaging facility.

Keywords: Hot Cell, Re-packaging, Concrete Wall, Lead Glass, Borosilicate Glass, Shielding

Occupational Dose Analysis for Fire Accident During NPP Decommissioning

Hyunjin Lee, Chang-Lak Kim*, Sang-Rae Moon, and Sun-kee Lee

Kepco International Nuclear Graduate School (KINGS)

*clkim@kings.ac.kr

Nuclear power plants decommissioning is planned to be started in middle of the 2020. It is necessary to develop safety evaluation and verification technology during decommissioning to ensure the safety of security monitoring measures and maintenance measures, appropriate emergency plans and preparations for decommissioning, and the use of proven engineering when establishing decommissioning plan. For this purpose, a nuclear power plant decommissioning plan is prepared in several stages before decommissioning. When a lifetime of a nuclear power plant has reached, it needs to be decommissioned and therefore operator company should submit decommissioning plans to the National Safety and Security Commission. And safety analysis should be included in this document and it is explained in chapter 6. According to the NSSC Notice No. 2021-10, it is largely divided into principles and standards, exposure scenarios, dose assessment, residual radioactivity, abnormal events, and risk analysis. When unexpected radiological accident is happened, both public and occupational dose analysis should be conducted. However, research on the former can be found easily on the other hands, research on the latter is not active.

In this paper, method of choosing scenarios of accidents during the decommissioning the nuclear power plants is briefly introduced. Accidents during nuclear power plants decommissioning cases in USA is chosen and its risk is evaluated by using risk matrix and ranked by AHP method.

During the decommissioning phases, varieties of radioactive waste is expected to be generated such as contaminated concrete and metal. On the other hand, Dry Active Waste (DAW) is generated and its amount is and its amount is 7,353 drums. Characteristic of DAW is highly flammable compared to concrete or metal. Moreover, depending on method of radioactive waste conditioning and type of radioactive nuclides, release rate of the nuclides varies. Thus this type of radioactive waste is critical to fire accidents and such accident can occur extra dose exposure which exceeds the guideline of the regulatory body to workers. Therefore, in this paper, occupational dose exposure during the fire accident is conducted.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Korea Foundation of Nuclear Safety (KOFONSD Project No. 2106023), Republic of Korea.

Keywords: Safety Analysis, Occupational Dose Analysis, Decommissioning, Scenario Selection

A Study on the Status of Multi-nuclide Removal System (ALPS) and Dilution/Discharge Process on Fukushima NPP Site

Myeonggi Cha, Gisu Lee, and Min Baek*

Pohang University of Science and Technology (POSTECH)

*baekmin@postech.ac.kr

From Fukushima nuclear disaster, as the water which is supplied by rain and groundwater flow into reactor building, contaminated water which contains radioactive nuclides is occurred. Although about 600 tons of contaminated water was generated at the early of accident, as the groundwater management system is developing, about 150 tons of contaminated water is generated now. Tokyo Electric Power Holdings (TEPCO) operate a multi-nuclide removal equipment which is called 'ALPS' and store purified water (ALPS treated water) in the Fukushima NPP site by tank. From 2023, the Japanese government decided to dilute the stored ALPS treated water and discharge it into the ocean to secure space on the site.

In this study, based on the data opened to the public by TEPCO, the current status of ALPS is investigated. The dilution and discharge process under conceptual design was investigated. In addition, the treatment capacity of ALPS was analyzed based on the radioactivity concentration data of 7 nuclides. And then, two points to be checked found. First, it was confirmed that the performance of ALPS temporarily decreased between 2015 and 2018 due to reduced replacement cycle of filter and absorbent. Second, it was confirmed that the ALPS treated water from specific ALPS still haven't satisfied the discharge limit for I-129, Sr-90, and Cs-137. In the case of Cs-137, about 1.7 times the radioactivity concentration was detected compared to the discharge limit. For I-129 and Sr-90, about 2.4 times and 2.1 times of radioactivity concentration was detected compared to the discharge limit.

From this study, some of the ALPS treated water are confirmed that the radioactivity concentration exceeds the discharge limit, and the treatment capacity of ALPS might be unstable depend on the ALPS operation such as replacement cycle. Therefore, before the discharging of contaminated water on 2023, it is necessary to inspect ALPS if it purifies contaminated water with reliability or not, and to secure the reliable evaluation method to measure radioactivity concentration.

Keywords: Fukushima, ALPS, Multi-nuclide Removal System, Discharge

6분과

방사선환경 및 안전(Poster)

The Mapping Program for the Display of Radiation Distribution Using Aerial Radiation Monitoring Data

Hwa Jeong Han¹, Sang Hun Shin², Hee Kwon Ku², Min Beom Heo², and Byung Gi Park^{1,*}

¹Soonchunhyang University

²FNC Technology

*byunggi@sch.ac.kr

Investigations and monitoring of environmental radiation are important for preventing expected accidents or for early detection of unexpected accidents, in nuclear facilities and the surrounding. In the event of an environmental radiation accident, it should be possible to identify and analyze the radiation-contaminated area. Therefore, a rapid radiation monitoring system is required for immediate response and necessary measures. In this study, the distribution of radiation mapping is performed on a contaminated area using 2-dimensional or 3-dimensional contour mapping techniques. The entire surrounding area can be understood at a glance by displaying the radiation contour line on the map of the measured area.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (No. 2020M2D2A2062538).

Keywords: Aerial Radiation Monitoring, Radiation Distribution, Contour Mapping, Aerial Radiation

Development and Field Tests of a Mobile Radiation Monitoring System Using Public Transportation

Jin Sik Choi, Jae Wook Kim, Han Young Joo, Jeong Yeon Lee, and Joo Hyun Moon*

Dankook University

*jhmoon86@dankook.ac.kr

Republic of Korea (ROK) is operating the Integrated Environmental Radiation Monitoring Network (IERNet) in preparation for a radioactive emergency based on Article 105 of the Nuclear Safety Act (Monitoring of Nationwide Radioactive Environment). 215 radiation monitoring posts are monitoring a wide area, but their location is fixed, so they can't cover areas where the post is not equipped around the Nuclear Power Plants (NPPs). For this, a mobile radiation monitoring system was developed using a drone or vehicle. However, there are disadvantages: it is performed only at a specific cycle, and an additional workforce is required. In this study, a radiation monitoring system using public transportation was developed to solve the above problems. Considering the range of dose rates from environmental radiation to high radiation doses in accidents, the detector was designed by combining NaI (TI) (in the low-dose area) and GM detector (in the high-dose area). Field test was conducted by installed on a city bus operated by Yeonggwang-gun to confirm the performance of the radiation monitoring system. As a result of the field test, it was confirmed that data is transmitted from the module to the server program in both directions. Based on this study, it will be possible to improve the radiation monitoring capability near nuclear facilities.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (Ministry of Science and ICT) (No. 2020M2D2A2062436) and the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) through financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2003019).

Keywords: Mobile Radiation Monitoring System, Public Transportation, GM Counter, NaI (TI) Detector, Radiological Preparedness

Spectroscopy- $H^*(10)$ Rate Conversion Function $G(E)$ for a Plastic Scintillation Detector

Jisung Hwang, Junhyeok Kim, Kilyoung Ko, Wonku Kim, Jaehyun Park, Gyohyeok Song, Sangho Lee,
and Gyuseong Cho*

Korea Advanced Institute of Science and Technology (KAIST)

*gscho@kaist.ac.kr

Dose-rate monitoring instruments are indispensable to protect workers from the potential risk of radiation exposure, and are commonly calibrated in terms of the ambient dose equivalent ($H^*(10)$), an operational quantity that is widely used for area monitoring. Plastic scintillation detectors are ideal equipment for dosimetry because of their advantages of low cost and tissue equivalence. However, these detectors are rarely used owing to the characteristics caused by low-atomic-number elements, such as low interaction coefficients and poor gamma-ray spectroscopy. In this study, we calculated the $G(E)$ function to utilize a plastic scintillation detector in spectroscopic dosimetry applications. Numerous spectra with arbitrary energies of gamma rays and their $H^*(10)$ were calculated using Monte Carlo simulations and were used to obtain the $G(E)$ function. We acquired three different types of $G(E)$ functions using the least-square and first-order methods. The performances of the $G(E)$ functions were compared with one another, including the conventional total counting method. The performance was evaluated using ^{133}Ba , ^{137}Cs , ^{152}Eu , and ^{60}Co radioisotopes in terms of the mean absolute percentage error between the predicted and true $H^*(10)$ values. In addition, we confirmed that the dose-rate prediction errors were within acceptable uncertainty ranges and that the energy responses to ^{137}Cs of the $G(E)$ function satisfied the criteria recommended by the International Commission.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MIST) (No. RS-2022-00154985).

Keywords: Spectrum-to- $H^*(10)$ Conversion, Plastic Scintillation Detector, $G(E)$ Function, Least-Square Method, First-order Method

Development of Virtual Gamma-Ray Energy Spectrum Simulator Program for Training

Yoomi Choi*, Young-Yong Ji, Sungyeop Joung, and Eunjoong Lee

Korea Atomic Energy Research Institute (KAERI)

*cym1031@kaeri.re.kr

Gamma spectrometry is one of the main analysis methods used to obtain information about unknown radioactive materials. In gamma-ray energy spectrometry, even for the same gamma-ray spectrum, the analysis results may be slightly different depending on the skill of the analyst. Therefore, it is important to increase the proficiency of the analyst in order to derive accurate analysis results. This paper describes the development of the virtual spectrum simulator program for gamma spectrometry training.

This simulator program consists of an instructor module and trainee module program based on an integrated server, in which the instructor transmits a virtual spectrum of arbitrarily specified measurement conditions to the students, allowing each student to submit analysis results. It can reproduce a virtual gamma-ray energy spectrum based on virtual reality and augmented reality technique and includes analysis function for the spectrum, allowing users to experience realistic measurement and analysis online. The virtual gamma-ray energy spectrum DB program manages a database including theoretical data obtained by Monte Carlo simulation and actual measured data, which are the basis for creating a virtual spectrum. The currently developed database contains data on HPGe laboratory measurement as well as in-situ measurements (ground surface, decommissioned facility wall, radiowaste drum) of portable HPGe detectors, LaBr₃(Ce) detector and NaI detector. The analysis function can be applied not only to the virtual spectrum, but also to the input measured spectrum. The parameters of the peak analysis algorithm are customizable so that even low-resolution spectra can be properly analyzed. The validity of the database and analysis algorithm was verified by comparing with the results derived by the existing analysis programs. In the future, the application of various in-situ gamma spectrometers will be implemented to improve the profiling of the depth distribution of deposited nuclides through dose rate assessment, and the applicability of the completed simulator in actual in-situ gamma spectrometry will be verified.

Acknowledgements

This work was supported by KOREA HYDRO & NUCLEAR POWER CO., LTD (No. G20IO02).

Keywords: Gamma Spectrometry, Virtual Gamma-Ray Energy Spectrum Simulator, Training program, Peak Analysis

Miniaturization and Applicability Evaluation of Collimating Device for the Self-Attenuation Correction in Low-Energy Gamma-Ray Spectrometry

Jun-Ho Lee* and Jong-In Byun

Korea Institute of Nuclear Safety (KINS)

*jhlee@kins.re.kr

In gamma-ray spectrometry for volume samples, the self-attenuation effect should be considered in the case of differences in chemical composition and density between the efficiency calibration source for quantitative analysis of sample and the sample actually measured. In particular, the lower the gamma-ray energy, the greater the gamma-ray attenuation due to the self-attenuation effect of the sample. So, the attenuation effect of low-energy gamma-rays in the sample should be corrected to avoid over- or under-estimation of its radioactivity. One of the most important factors in correcting the self-attenuation effect of the sample is the linear attenuation coefficient for the sample, which can be directly calculated using a collimator. The larger the size of the collimator, the more advantageous it is to calculate the linear attenuation coefficient of the sample, but excessive size may limit the use of the collimator in a typical environmental laboratory due to its heavy weight. Therefore, it is necessary to optimize the collimator size and structure according to the measurement environment and purpose.

This study is to optimize a collimator that can determine the effective linear attenuation coefficient of low-energy gamma-rays, and verify its applicability. The overall structure of the designed collimator was optimized for gamma-ray energy of less than 100 keV and cylindrical plastic bottle with diameter of 60 mm and a height of 40 mm. The materials of optimized collimator consisted of tungsten. Acryl and acetal were used to form the housing of the collimator, which fixes the central axis of the bottle, collimator and point-like source. In addition, using the housing, the height of the tungsten is adjusted according to the height of the sample.

For applicability evaluation of the optimized collimator, IAEA reference material in solid form were used. The sample was filled in the bottle with heights of 1, 2, 3 and 4 cm respectively. Using the collimator and point-like source of ^{210}Pb (46.5 keV), ^{241}Am (59.5 keV), and ^{57}Co (121.1 keV), the linear attenuation coefficient and the radioactivity for the samples were calculated.

As a result, to calculate the linear attenuation coefficient using the optimized collimator, a relatively high sample height is required. However, the optimized collimator can be used to determine the linear attenuation coefficients of low-energy gamma-rays for the self-attenuation correction regardless of the sample height. It is concluded that the optimized collimator can be useful to correct the sample self-attenuation effect.

Keywords: Gamma-Ray Spectrometry, Low-Energy Gamma-Emitting Radionuclides, Self-Attenuation Effect, Collimator, Linear Attenuation Coefficient

Feasibility Study on Gamma Energy Spectroscopy of Bismuth Nanoparticles-loaded Plastic Scintillator for Portal Monitoring Using Monte Carlo Simulations

Taeseob Lim¹, Siwon Song¹, Jae Hyung Park¹, Jinhong Kim¹, Seunghyeon Kim¹, Hyungi Byun^{1,2},

Seokhyeon Jegal¹, and Bongsoo Lee^{1,*}

¹Chung-Ang University (CAU)

²FNC Technology

*bslee@cau.ac.kr

Plastic scintillators can be used to find radioactive sources for portal monitoring due to their advantages such as faster decay time, non-hygroscopicity, relatively low manufacturing cost, robustness, and easy processing. However, plastic scintillators have too low density and effective atomic number, and they are not appropriate to be used to identify radionuclides directly. In this study, we devise the radiation sensor using a plastic scintillator with holes filled with bismuth nanoparticles to make up for the limitations of plastic materials. We use MCNP (Monte Carlo N-particle) simulating program to confirm the performance of bismuth nanoparticles in the plastic scintillators. The photoelectric peak is found in the bismuth-loaded plastic scintillator by subtracting the energy spectrum from that of the standard plastic scintillator. The height and diameter of the simulated plastic scintillator are 3 and 5 cm, respectively, and it has 19 holes whose depth and diameter are 2.5 and 0.2 cm, respectively. As a gamma-ray source, Cs-137 which emits 662 keV energy is used. The clear energy peak is observed in the subtracted spectrum, the full width at half maximum (FWHM) and the energy resolution are calculated to evaluate the performance of the proposed radiation sensor. The FWHM of the peak and the energy resolution are 61.18 keV and 9.242% at 662 keV, respectively.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grants funded by the Korean government (MSIT) (No. 2020M2D2A2062457, 2020M2D8A2066404, 2022M2D4A1084440) and the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korean government (MOTIE) (No. 20201520300060).

Keywords: Plastic Scintillator, Bismuth Nanoparticles, Monte Carlo N-particle Simulation, Subtraction Method

Design Improvement on Reducing Angular Dependence of Gamma Probe for Detecting Radioactive Contamination

Sanggeol Jeong, Giyoon Kim, Jaeyeong Jang, Heejun Chung, and Myungsoo Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*myungsookim@kinac.re.kr

When the nuclear accident like the Fukushima is occurred, it is required to immediately determine the location of radioactive materials and their activities. Various studies related the unmanned technique to detect and characterize the contaminated area have been conducted. The Korea Institute of Nuclear Nonproliferation and Control (KINAC) has developed a new gamma detection system which consists of nine probes using a silicon photomultiplier (SiPM) and plastic scintillator. The probe is the small gamma detector designed to be carried and dropped near the accident area by the unmanned aerial vehicle.

In this paper, we developed the improved design related to the angular dependence of the radioactive contamination detection system with the purpose of increasing the detection efficiency. The detection efficiency, radiation shielding and back-scattering varies depending on the direction of incidence of radiation because the probe has vertical structure of consisting scintillator, photomultiplier, and electric circuits. That is, when the experimental conditions are same except the direction of gamma probe, the result of measurements is different. It causes errors in measuring the radioactivity and location of the radioactive source. Since the direction of the probe is arbitrarily determined during the deployment of the probe through the unmanned aerial vehicle, it is considered changing the design of the scintillator from a conventional $1.0'' \times 1.0'' \Phi$ cylindrical shape to a $1.0'' \Phi$ spherical shape. In case of using the spherical scintillator, it is confirmed that angular dependence was reduced through MCNP simulation. The difference in the measurement depending on the direction of the probe could be reduced through additional structure design.

Finally, we hope that the developed detection system which has the probes with spherical shape of scintillator can measure the radioactivity and location of the radioactive source in a range of about $100 \times 100 \text{ m}^2$ by measuring for at least 5 minutes. The field test at Fukushima area will be carried out with JAEA members in order to prove the feasibility of the new system.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1905010).

Keywords: Gamma Probe, Plastic Scintillator, Angular Dependence, Nuclear Accident, Denuclearization

Minimization for Temperature Dependence of the SiPM Adapting Count Rate Compensation and Operation Sequence

Giyeon Kim, Sanggeol Jeong, Jaeyeong Jang, Heejun Chung and Myungsoo Kim*

Korea Institute of Nuclear Nonproliferation and Control (KINAC)

*myungsookim@kinac.re.kr

In emergency situations such as nuclear accidents or terrorism, radioactive and nuclear materials can be released by some environmental reasons such as the atmosphere and underground water. To secure the safety of human beings and to respond appropriately emergency situation, it is required to designate high and low dose rate regions in the early stages by analyzing the location and radioactivity of sources through environmental radiation measurement.

This research team has developed a small gamma probe which is featured by its geometrical accessibility and higher radiation sensitivity than other drone detectors. A plastic scintillator and Silicon Photomultiplier (SiPM) were applied to the probe to optimize the wireless measurement condition. SiPM has a higher gain (higher than 10^6) and lower operating voltage (less than 30 V) compared to a general photodiode. However, the electronic components in the SiPM are sensitively affected by temperature, which causes the performance degradation of the SiPM. As the SiPM temperature increases, the breakdown voltage (V_{BD}) of the SiPM also increases, so the gain must be maintained by applying the appropriate V_{BD} . Therefore, when the SiPM temperature increases while the V_{BD} is fixed, the gain decreases. Thus, the signal does not exceed the threshold voltage (V_{TH}) and the overall count is reduced. In general, the optimal gain is maintained by cooling the SiPM or through a temperature compensation circuit. However, in the developed system, the hardware correction method such as cooling or temperature compensation circuit cannot be applied.

In this study, it was confirmed that the count decreased by up to 20% according to the increase in the temperature of the SiPM when the probe was operated at room temperature (26°C). We propose methods to calibrate the total count without cooling device or compensation circuit. After operating the probe at room temperature, the first measured count is set as the reference value, and the correction factor is derived using the tendency of the count to decrease as the temperature increases. In addition, since this probe is used for environmental radiation monitoring, periodic measurements are more suitable than continuous measurements. Therefore, the temperature of the probe can be maintained by adding a power saving interval to the operation sequence of the probe. These two methods use the operation sequence and measurement data, respectively. Thus, it is expected to be the most effective method for the current system where the temperature compensation through hardware is not possible.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1905010).

Keywords: Denuclearization, Gamma Probe, Plastic Scintillator, SiPM, Detection Efficiency

Scintillating Detection of Fe-55 for Radioactive Aerosol Monitoring

Jae Hyung Park, Siwon Song, Seunghyeon Kim, Jinhong Kim, Taeseob Lim, Seokhyeon Jegal, and Bongsoo Lee*

Chung-Ang University

*bslee@cau.ac.kr

As the decommissioning and decontamination (D&D) of nuclear power plants (NPPs) has actively proceeded worldwide, the management of radiation exposure of workers has become more critical. Radioactive aerosol is one of the main causes of worker exposure, contributing to internal exposure by inhalation. It occurs in the process of cutting radioactive metal structures or melting radioactive wastes during D&D, and its distribution varies according to decommissioning strategies and cutting methods. Among the dominant radionuclides in radioactive aerosols, Fe-55 is known to be the most abundant.

Fe-55, which decays by electron capture, is classified as a difficult-to-measure (DTM) radionuclide because its emitted X-rays have too low energy to measure directly from outside of the container. Generally, for measuring DTM nuclides, the liquid scintillation counting (LSC) method and the scaling factor (SF) method are used. However, these methods are not suitable for continuous monitoring of the D&D workplace due to the necessity of sampling and additional analysis. The radiation measurement system that can directly measure the radionuclides collected at the aerosol filter could be more useful.

In this study, as preliminary research on developing the radioactive aerosol monitoring system, we fabricated a gamma-ray spectrometer based on a NaI (TI) scintillator and measured the energy spectrum of Fe-55. A beryllium window was applied to the scintillator for X-ray transmission, and the Fe-55 check source was directly attached to the scintillator assuming that the aerosol filter was equipped. 5.9 keV photopeak was clearly observed and the energy resolution was estimated as 44.10%. Also, the simultaneous measurement with Cs-137 was carried out and all the peaks were measured.

Acknowledgements

This research was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korean government (MOTIE) (No. 20201520300060) and the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) (No. 2020 M2D2A2062457, 2020M2D8A2066404, 2022M2D4A1084440).

Keywords: Decommissioning, Radioactive Aerosol, Fe-55, Radiation Monitoring, Energy Spectrum

Assessment of Cosmic-Ray Contribution on Aerial Survey Using Large Volume NaI (Tl) Detector

Jun Ha Kim and Jong In Byun*

Korea Institute of Nuclear Safety (KINS)

*k975bji@kins.re.kr

This study was performed to assess the cosmic-ray effect caused by altitude in the aerial gamma-ray measurement. For the gamma-ray measurement experiment by altitude, the aerial survey system composed of four 4×4×16 inches large volume NaI (Tl) detectors was used. The aerial survey system was installed in a rotor-craft to stably keep its flight altitude and position. In addition, in order to avoid time-dependent shielding effects with the amount of fuel, a rotor-craft of which the fuel tank is not located beneath the cabin floor was selected. In this study, the ROI (Region Of Interest) was set to the 3~6 MeV range to assess the cosmic-ray contribution to the gamma-ray spectrum that could ignore the contribution of the dominant natural radionuclides.

The gamma-ray spectra measured inside and outside of the rotor-craft on the ground were compared to evaluate the shielding effects of the aircraft body. As a result, the count rate of the ^{40}K photo peak was decreased by about 10% when measuring the inside compared to the outside. On the other hand, the total count rate of the 3~6 MeV region was decreased by about 0.7% under the same condition. Therefore, the aircraft body effect was insignificant in 3~6 MeV region considering the relative uncertainty of 0.04~0.78% (1σ).

In addition, the count rate in the 3~6 MeV range according to altitude was evaluated to assess the cosmic-ray effect. In order to evaluate the change in the ROI count rate according to the altitude, the gamma-ray spectrum was measured in the range of 300~2,000 m above the sea to avoid the effect of terrestrial radiation. As a result, the relationship between altitude and count rate in the 3~6 MeV range showed a high correlation with the R^2 value of 0.99, when the approximate equation was derived in the form of a quadratic polynomial.

Also, the count rate of 3~6 MeV at 50~500 m above the ground was estimated using the correlation equation, and this value was compared with the measured count rate. As a result of comparing the average value of estimated count rate and measured count rate, the relative difference is less than 2%. Considering the relative uncertainty of 0.78~4.11% (1σ), it was possible to evaluate the count rate of the 3~6 MeV region relatively accurately. The results of this study could be used for further study on background dose corrections in aerial survey.

Keywords: Aerial Survey, Radiation Monitoring, Cosmic ray, NaI (Tl) detector

Evaluation of ^{40}K Concentration in Samples Containing High Concentration of ^{232}Th

Eun Ji Park*, Sung Han Lee, Young Hwan Park, Su Bin Heo, Seong Gyu Lim, Dong Cheol Jang,

and Seung Beom Seo

HANIL Nuclear

*eunji03@hanilnuclear.co.kr

In 2018, media reports raised issues related to radon released from building materials used as finishing materials in apartment houses. Accordingly, related ministries recommended not to use materials with a radiation index value exceeding 1.

In order to calculate the radioactivity index, not only ^{226}Ra producing radon (^{222}Rn) but also ^{232}Th and ^{40}K radioactivity concentrations are required. To determine the concentration of the radionuclide, ^{40}K is measured by a single gamma ray of 1,460.8 keV. And the ^{228}Ac used to measure ^{232}Th mainly utilizes gamma rays of 911.2 keV. However, ^{228}Ac does not appear as a single peak unlike ^{40}K , and appears as multiple peaks at various energies. Among them, gamma rays are emitted at an intensity of 0.83% at 1,459.2 keV, which is likely to interfere with ^{40}K . Therefore, what is actually measured at 1,460.8 keV is theoretically a compound peak of ^{40}K and ^{228}Ac . Because the probability of emission at 1,459.2 keV (0.83%) is low, a low concentration of ^{232}Th will result in little ^{40}K radioactivity error. However, samples containing a high concentration of ^{232}Th overestimate the ^{40}K radioactive concentration, so correction is required.

In this study, the IAEA standard substance (IAEA-RGTh-1) containing ^{232}Th of actual high concentration was analyzed, and the results of the analysis without correction of ^{40}K were compared and verified. As the ^{40}K correction method, the 911.2 keV gamma-ray of ^{228}Ac was used as the reference peak to separate the peak of ^{228}Ac (1,459.2 keV) from the ^{40}K (1,460.8 keV) mixed peaks. And, the coefficient value obtained by subtracting the peak of ^{228}Ac (1,459.2 keV) from the ^{40}K (1,460.8 keV) mixed peak was set to a pure peak of ^{40}K and the radioactivity concentration was calculated therefrom. As a result of calculating the IAEA-RGTh-1 reference material without correction, it was confirmed that the ^{40}K value was overestimated by about 38 times.

If a measurement beyond the MDA of ^{40}K is generated by ^{228}Ac radioactivity because the ^{40}K correction constant is not applied, there is an error in determining that there is ^{40}K radioactivity. However, even if ^{40}K radioactivity is overestimated due to the high concentration of ^{232}Th , the degree to which this effect contributes to the radioactivity index is very small. However, as an analyst, ^{40}K radioactivity correction should be made for more accurate analysis.

Keywords: Thorium, Potassium, Environmental Radioactivity, Correction of ^{40}K

Machine Learning Approach to Radiation Exposure Doses From Consumer Products Containing Naturally Occurring Radioactive Materials

Juyoul Kim^{1,*} and Gi Young Han²

¹Kepeco International Nuclear Graduate School (KINGS)

²Korea Institute of Nuclear Safety (KINS)

*jykim@kings.ac.kr

Since radon was detected in mattresses of famous bed furniture brands in 2018, the nuclear safety and security commission (NSSC) announced the radiation safety management act in April 2021 to protect the public health and environment. This act stipulates the safety management of radiation that can be encountered in the natural environment such as the notification of radioactivity concentration of source materials, process by-products, the installation and operation of radioactive monitors. In this study, a model was established to predict radioactive exposure dose from radioactive materials such as radon and uranium detected in consumer products such as bed mattresses, pillows, shower, bracelets and masks in order to identify major radioactive substances that largely affect the exposure dose. A period of seven years from 2014 to 2020 was investigated for the source materials and exposure doses of consumer products containing naturally occurring radioactive materials (NORMs). We analyzed these using machine learning models such as classification and regression tree (CART), Random Forest and TreeNet. Index development and verification were performed to evaluate the predictive performance of the models. Overall, predictive performance was highest when Random Forest or TreeNet was used for each consumer product. Thoron had a great influence on the internal exposure dose of bedding, clothing and mats. Uranium had a great influence on the internal exposure dose of other consumer products except whetstones. When the number of data is very small or the missing value rate is high, it is difficult to expect accurate predictive performance even with machine learning techniques. If we significantly reduce the missing value rate of data or use the limit of detection value instead of missing values, we can build a model with more accurate predictive performance.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2003015). This research was also supported by the 2022 Research Fund of the KEPCO International Nuclear Graduate School (KINGS), the Republic of Korea.

Keywords: Naturally Occurring Radioactive Material, Consumer Product, Radiation Safety Assessment, Machine Learning Approach

Analysis on the Country-specific Graded Approaches to Exemption and Clearance of NORM

Zuhee Woo*

Korea Institute of Nuclear Safety (KINS)

*k698wzh@kins.re.kr

EU BSSD (2013/59/EURATOM) requires that NORM involving activities are managed within the same regulatory framework as other practices causing exposure to ionizing radiation, and a graded approach to regulatory control needs to be applied based on radiation risk. The graded approach is applied to all sources of radiation subject to regulation. However, it should be adopted for industrial activities involving NORM. In the case of the regulatory control of NORM, the application of the concepts of exemption and clearance plays a key role. Most of the European countries have adopted the new EU BSSD general exemption/clearance levels, 1 Bq/g for U/Th series and 10 Bq/g for K-40. In addition to the general exemption or clearance criteria, some countries have implemented the graded regulatory approach, adopting more restrictive criteria as a specific exemption or clearance based on the risk of 1 mSv/y for workers and 0.01-0.3 mSv/y for public.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106009).

Keywords: NORM, Exemption, Clearance, Graded Approach

Distribution of Naturally Occurring Radioactive Materials in Groundwater of Jeju Island

Sohyeon Lim^{1,2}, Chung-Hun Han¹, and Hee-Jung Im^{1,*}

¹Jeju National University (JNU)

²Water Supply and Sewage Treatment Headquarters

*sh2952@korea.kr

This study presents distribution of naturally occurring radioactive materials in groundwater in Jeju island. Radon (^{222}Rn) and potassium (^{40}K) concentrations were performed by using RAD H₂O of RAD7 and 940 Professional IC Vario, respectively. In addition, the activities of uranium and thorium nuclides were analyzed by ICP-MS. All of the groundwater samples were collected from 29 sites from August to October 2022. The radon concentrations in groundwater were in the range of 0 to 60 Bq L⁻¹, and there was no groundwater exceeding the range of 148 Bq L⁻¹ proposed by the US EPA. The distribution of uranium in groundwater varied from 0 to 0.6 µg L⁻¹ and did not exceed 30 µg L⁻¹, thresholds indicated by the US EPA.

Keywords: ^{222}Rn , ^{40}K , Uranium, Thorium, Groundwater, Jeju

Property Analysis of Aerosol Containing NORM in Cement Manufacturing Industry for Radiological Risk Assessment

Jong Hyeok Park, Byung Min Lee, Seong Yeon Lee, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

Cement is widely used as representative industrial material. In Korea, about 50 million tons of cement are consumed every year. In the manufacture of cement, raw materials containing NORM such as fly ash and bauxite are used. Therefore, the workers can be subjected to radiation exposure. The major exposure pathway in NORM industries is internal exposure due to inhalation of aerosol. Internal radiation dose due to aerosol inhalation varies depending on physicochemical properties of the aerosol. Therefore, the objective of this study was to investigate aerosol properties influencing inhalation dose in cement industries. In this study, aerosol properties were measured for two cement manufacturers. A particulate size distribution and concentration at various processing areas in cement manufacturing industries in Korea were analyzed using a cascade impactor. The mass density of raw materials and byproducts were measured using pycnometer. Shape of particulates was analyzed using SEM. The radioactivity concentration of Ra-226, Ra-228 for U/Th decay series was measured using HPGe. Particulate concentration by size was distributed log-normally with maximum at particle size about 7.2 μm in manufacturer A and 5.2 μm in manufacturer B. The mass density of fly ash and cement were 2.3 ± 0.06 , 3.2 ± 0.02 g/cm^3 respectively in manufacturer A. In manufacturer B, the mass density of bauxite and cement were 3.4 ± 0.02 , 2.9 ± 0.01 g/cm^3 respectively. The shape of particulates appeared as spherical shape in manufacturer A and B regardless of sampling area. Thus, a shape factor of unity could be assumed. The radioactivity concentrations of Ra-226, Ra-228 were 82 ± 9 , 82 ± 8 Bq/kg for fly ash, and 25 ± 4 , 23 ± 3 Bq/kg for cement in manufacturer A. In manufacturer B, the radioactivity concentrations of Ra-226, Ra-228 were 344 ± 34 , 391 ± 32 Bq/kg for bauxite, and 122 ± 13 , 145 ± 12 Bq/kg for cement. The radioactivity concentrations of Ra-226, Ra-228 in cement were less than raw materials such as fly ash and bauxite. It is because the dilution of the radioactivity concentration occurred during mixing with other raw materials in cement production process. This study results will be used as database for accurate dose assessment due to airborne particulate inhalation by workers in cement industries.

Acknowledgements

This work was supported through the KoFONS using the financial resource granted by NSSC (No. 2105035).

Keywords: NORM, Aerosol Property, Cement Manufacturing, Dose Assessment

Disposal of a Radioactive Concrete Waste Containing Uranium by the Crush-Sieving Method

Jeong-bo Yang*, Jeong-myeong Kim, Jung-hwan Hong, Yeong-jong Ju, and Jung-suk Park

KEPCO Nuclear Fuel (KEPCO NF)

*yangjb@knfc.co.kr

Concrete waste generated in the result of dismantling a concrete structure in a radiation control area and refractory brick waste generated from uranium pellet sintering furnace are surface-contaminated by uranium particle of which the enrichment is below 5%. These wastes are hard to decontaminate so it was necessary to develop the process for its disposal.

So, we developed the Process Control Plan (PCP) for disposal of radioactive concrete waste describing a whole sequence of disposal and inspecting procedures based on the KNF Radioactive Waste Quality Assurance Plan (KN-WQAP) established in 2021. Based on the PCP, we crushed the concrete waste by jaw-crusher. Then we sieved the crushed concrete waste and removed the particle of which size is below 0.3 mm, using sieve-vibrator where the 0.3 mm mesh-sized sieve is installed inside.

Before conducting the crush-sieving method based on the PCP, we conducted Process Control Assessment (PCA) based on the KN-WQAP. The purpose of the PCA is to check whether the output of the process satisfies the Acceptance Criteria of Korea Radioactive Waste Agency (KORAD) so that we could confirm the validity of the PCP. The evaluation item of the PCA is a particulate size verification test. The test is passed only if the component ratio of a particle size below 0.2 mm is less than 15% and the particle size below 0.01 mm is less than 1%. The very first 3 drums passed the test, so we began applying the PCP to whole target drums.

In the process of conducting the crush-sieving method in earnest, qualified inspectors based on KN-WQAP participated conducting sampling, measuring and checking whether a foreign material was included. They tested samples and packaged drums regarding 5 spheres of general, radiological, physical, chemical and biological characteristic.

KNF disposed concrete and refractory brick waste by the crush-sieving method so that KNF could take over 100 drums to KORAD in 2021. But, it is needed to be improved that a dust size below 0.3 mm is generated as a secondary waste which needs to be solidified for the final disposal and the work environment is not good enough because of the dust.

Keywords: Radioactive Concrete Waste, Refractory Brick, Radioactive Waste Disposal, Crush-sieving Method

Optimization of Design of a D₂O-moderated ²⁵²Cf Source for Personal Dosimetry Performance Test

Hyeongjin Kim*, Yuho Weon, and Munhyung Cho

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*kim.hj1222@khnp.co.kr

Among the test categories of the personal dosimetry performance test in Korea, the reference neutron radiation field used for the mixed neutron-photon radiation field is generated by a D₂O-moderated ²⁵²Cf source. There are some differences depending on the standards, D₂O-moderated ²⁵²Cf source consists of the ²⁵²Cf source surrounded by the D₂O sphere with a diameter of 30 cm, covered with a Cd shell of thickness approximately 0.051 cm ~ 0.1 cm. In order to optimize the design of the D₂O sphere and establish the neutron radiation field for the personal dosimetry performance test in Central Research Institute of KHNP, the neutron spectra have been simulated by MCNP 6.2 code by design conditions and evaluated dose conversion coefficients.

In the consideration of neutron irradiation facility, the basic design structure was determined a D₂O sphere with a diameter of covered with a Cd shell and a cylindrical well is in the middle of sphere. Neutron source transfer tube is inserted into this well-shaped structure and neutron source was withdrawn from the tube. And by changing the following design conditions in detail, the neutron spectra were evaluated; 1) the entire diameter of the D₂O sphere (with or without the diameter of 7.5 cm of well-shaped structure) 2) the location of the neutron source (distance from D₂O) 3) thickness of Cd shell 4) purity of D₂O. As a source spectrum, the spectrum of bare-²⁵²Cf recommended by ISO 8529-1 was adopted and spectra were tallied using F4 tally at a distance of 120 cm from the neutron source. Finally, the fluence to dose conversion coefficients were calculated using the simulated spectra.

As a result of the evaluation, in case that an entire diameter of the D₂O sphere with a diameter of the source tube is 37.5 cm, the fluence to dose conversion coefficient was evaluated to about 4.4% lower than an entire diameter of the D₂O sphere is 30 cm. And in case that the distance between the D₂O and the top of the neutron source was about 3.75 cm which is a radius of well-type structure, it was evaluated to about 1.4% larger than the distance was about 1 cm, and when the thickness of Cd was 0.1 cm, it was evaluated to 0.8% larger than when it was 0.051 cm. Finally, when the purity of D₂O was 99.99%, it was evaluated 1.5% lower than when it was 99%. Except for the diameter of D₂O sphere, the differences on the other conditions are acceptable considering the uncertainty of the simulation.

Therefore, the design of D₂O-moderated ²⁵²Cf source was determined by considering source integrity, economic perspectives, and dose conversion coefficient given in ISO 8529: a D₂O sphere with an entire diameter of 37.5 cm, filled with above 99% purity of D₂O, and covered with a cadmium thickness of 0.1 cm. The fluence to dose conversion efficient was evaluated as 110.10 pSv cm² for ambient dose H*(10), 115.21 pSv cm² for personal dose Hp(10) respectively.

Keywords: D₂O-moderated ²⁵²Cf source, Design, MCNP Code, Dose Conversion Coefficients

Review of Environmental Radiation Monitoring Strategy in Nuclear Emergency

Bongseok Kim*

Korea Atomic Energy Research Institute (KAERI)

*kbs@kaeri.re.kr

When the leakage of radioactive material or radiation to the environment or a concern, it is important to accurately understand the impact on the environment. Therefore, environmental effects evaluation using modeling based on meteorological data and source-term data is carried out, or environmental radiation monitoring which is an emergency response activity that directly measures dose is performed.

As lessons learned from the Fukushima accident, environmental effects evaluation and modeling cannot utilize during the emergency and decision-making process for protective action for the public. Thus, rapid environmental radiation monitoring is required. In Korea, when an emergency is issued at a nuclear facility, urgent environmental radiation monitoring is conducted based on the national nuclear emergency preparedness and response plan, which can provide important information for decision-making on public protective actions. A review of strategies for urgent environmental radiation monitoring is important in performing efficient emergency responses.

The main purpose of urgent environmental radiation monitoring is to gather data for decision-making on public protective actions to minimize the damage from the accident. For effective data collection and distribution, support from the national and local government and local public organizations and radiation expertise groups, and nuclear facility licensee are required. In addition, an emergency environmental radiation monitoring manual is required to immediately perform environmental monitoring in an emergency situation. The manual for emergency monitoring should include the activities to be conducted according to the phases of the emergency. The phases of the emergency are divided into pre-leakage, post-leakage, intermediate, and recovery.

The reasons for establishing strategies are government and public information, the implementation of urgent population protection countermeasures, predicting and tracking plume trajectory, and detection of any release, the protection of emergency and recovery workers, the implementation of agricultural countermeasures and food restrictions, the implementation of intermediate- and recovery-phase countermeasures, contamination control. Besides meteorological data, ambient dose rate and dose, airborne radionuclide concentration, environmental deposition, food, water, and environmental contamination, individual dose, and object surface contamination data are also required for making information for the public.

Keywords: Environmental Radiation Monitoring, Nuclear Emergency, Public Protective Action, Emergency Response

The Results of Performance Test for Whole Body Counters in Nuclear Power Plants

Si Young Kim*, Seok-Ju Hwang, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*siyoung.kim@khnp.co.kr

It is likely to occur internal exposure for workers in Nuclear Power Plants (NPPs) due to the intake of radionuclide. To assess the internal exposure dose the measurement of activity for remain radionuclide is necessary. The Whole Body Counters (WBCs) are commonly used for measurement of remain radionuclide activity in human body. Korea Hydro & Nuclear Power Co., Ltd. (KHNP) conduct performance test of WBCs in all NPPs for every year to confirm the performance of 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 2022 for all WBC is meet the ANSI N13.30 criteria. And the RMSE values are confirmed from 0.01 to 0.23. This means that the residual radioactivity measurement results using WBC are reliable.

Keywords: Internal Exposure, Whole Body Counter, Performance Test, ANSI N13.30

A Proposal for the Development Digital Twin in the Decommissioning Nuclear Power Plant Project

Sanghyun Lee*, Hyung-woo Seo, Gi-lim Kim, Jihwan Yu, and Hyein Kim

Korea Hydro & Nuclear Power (KHNP)

*12zero514@khnp.co.kr

The decommissioning of a nuclear power plant is a project that consists of several stages, and various technologies are applied when performing various tasks at each stage. And it is essential to secure safety and economic feasibility. As the paradigm has changed due to digital transformation in various industries, digitalization is applied to the life cycle of nuclear power plant from construction, operation and decommissioning project. Element technologies are being developed for decommissioning plan establishment, process design, econtamination method, decommissioning work process, waste management, environmental monitoring and radiation dose simulation.

The utilization of digital twin in the decommissioning stage is classified into three categories.

① Process Monitoring (decommissioning work procedure, work progress (plan/actual), real-time work status and etc.)

② Facility Monitoring (real-time sensing and video data monitoring, decommissioning SSCs information, work alarm and etc.)

③ Safety Monitoring (work safety, radiation exposure, fire monitoring, work risk and etc.)

A system suitable for the decommissioning stage and work should be developed in consideration of the target of use, development function, and when to create data according to the purpose of the system. Simulation module according to user purpose should be provided. In addition, data-base management should be performed according to the decommissioning characteristics in consideration of the data associated with the existing operating system. The system to be developed should support the project management to comply with the domestic standards and regulations to be determined in the future. This will improve the competitiveness of domestic and foreign markets.

Keywords: Decommissioning, Digital Twin Strategy, Project Management

An Improvement of Dimensioning Machine for Irradiated Materials in Hot Cell

Yunmock Jung*, Seungje Baik, Youngjun Kim, and Hyojik Lee

Korea Atomic Energy Research Institute (KAERI)

*jym@kaeri.re.kr

The dimensioning machine installed in the hot cell has been used for 20 years. It has been used for a long time so it was often malfunction due to aging and radiation. In addition, some parts of apparatus were discontinued and there were a lot of problems in maintenance and repair. In the old measuring system, the operator's subjectivity was much involved. The process of control the focal length (distance between lens and specimen) by operator's sense is a good example. The existing dimensioning machine was the Kh-7700 of Hirox Co., Ltd. As the equipment had been used for a longtime, additional utilities such as jigs, lighting module and servo motors have been customized and used. The same company's apparatus was selected for the reasons that it did not need to manufacture a new utility so it could save the cost of radioactive waste disposal for existing utilities and its radiation resistance which has been used for 20 years in radiation environment.

Lighting, standing, stage, controllers, cables and so on had been customized to provide remote control in hot cell. The installation was completed in March of this year in hot cell and has been successfully used until now. Through the improvement of dimensioning machine, an auto-focusing and multi-focusing were available. Therefore, they made it possible to produce high quality data and improve the accuracy of data. And this research could be a good example of how hot cell devices can be built and customized to achieve remote control.

Keywords: Dimensioning Machine, Hot Cell, Remote Control, Hirox

Preliminary Analysis of Uncertainties in the Calculated Value of Radioactivity for Airborne Discharges From NPPs in Batch Mode

Seok Hee Chung and Jae Hak Cheong*

Kyung Hee University

*jhcheong@khu.ac.kr

Discharge limits for nuclear power plant gaseous effluents are presented as dose constraints or on the basis of radioactivity or radioactivity concentration. Accordingly, the operator evaluates the amount of radioactive material discharged from a specific nuclear power plant to the environment and periodically reports them to regulators. Multi-step sampling and analysis and calculation are performed during the radioactivity evaluation process of radioactive effluent, and the uncertainty generated in each step causes the uncertainty of the final radioactivity. Considering that the purpose of evaluating radioactivity discharged from nuclear power plants to the environment is to verify the satisfaction of discharge limits and safety margins, it is necessary to accurately evaluate the discharged radioactivity as much as possible, understanding of the uncertainty contained in the reported value of radioactivity and efforts to reduce it.

In this study, modelling of the radioactivity evaluation procedure in gaseous effluent discharged as batch mode from nuclear power plant has performed, a generalized framework was established to evaluate the uncertainty based on ISO/IEC Guide 98-3 (GUM: 1995) involved in the whole process, and the uncertainty contained in the calculated radioactivity of each radionuclide (group) was evaluated and its characteristics. In addition, through probabilistic evaluation, the actual probabilistic distribution and statistical characteristics of radioactive effluent releases reported as a single value were confirmed.

As a result, the range of values expected to be included in the confidence level of approximately 95% of the distribution of values for radioactivity in a gaseous effluent discharged as a batch mode from nuclear power plant was calculated. And, the priority of each input parameter turned out to be (1) gaseous waste volume, (2) sample bottle volume, and (3) measured radioactivity of the sample.

In addition, the probability distribution of the radioactivity was simulated by Monte Carlo method. As such, the mean, minimum, and maximum values in confidence level of 95% were obtained, and they were reasonably matched the calculated value within 5% deviation. It was shown that radioactivity to the environment, which has been reported as a single value, has a specific probabilistic distribution form.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS), granted financial from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 2003017).

Keywords: Radioactive Discharge, Gaseous Effluent, Batch Mode, Radioactivity, Uncertainty, Probability Distribution

Review on Evaluation Method by Direct Radiation Exposure in ODCM

Seok-Ju Hwang*, Deuk-man Kim, and Cheon-Woo Kim

Korea Hydro and Nuclear Power (KHNP) Central Research Institute (KHNP-CRI)

*hsjhhhhh@khnp.co.kr

In accordance with the notification of the Nuclear Safety and Security Commission (NSSC), environmental impact assessments around nuclear power plants are conducted annually and the results are disclosed to the public. The effects of direct radiation exposure from nuclear power plants as well as liquid effluents and gaseous effluents are taken into consideration in the evaluation of dose calculation for residents.

In the United States, regulatory guidelines on direct radiation exposure are described in Reg. Guide 4.1, and the effects of direct radiation are evaluated through regulatory guidelines in Korea. We are going to review optimal evaluation method by reviewing the direct radiation exposure evaluation method currently being conducted in domestic nuclear power plants and the direct radiation exposure evaluation method in overseas nuclear power plants such as in the United States.

Keywords: Off-site Dose Calculation Manual, Liquid Effluent, Gaseous Effluent, Direct Radiation Exposure

Review of the Linkage Between Accident Management Plan and Radiation Emergency Plan

Juyoul Kim^{1,*} and Hyeong-Ki Shin²

¹Kepeco International Nuclear Graduate School (KINGS)

²Korea Institute of Nuclear Safety (KINS)

*jykim@kings.ac.kr

The correlation between accident management plan and radiation emergency plan of Shin-Kori Units 3 and 4 was compared and analyzed from the point of view of the adequacy of facilities, equipments, organization and manpower which are necessary for the related emergency response. It was found the equipment of accident management plan and emergency response facility of radiation emergency plan had different technical contents and scope of application, so there was no risk of mutual conflict and overlapping functions. However, since the accident impact assessment code in accident management plan and computer program of radiation emergency plan were different, it was necessary to ensure the agreement or linkage of the evaluation between them. When a radiation emergency is issued in accident management plan, the composition and mission of the accident response organization were mostly consistent with the contents of the radiation emergency plan, but some corrections and improvement items were identified. Accident management plan specified that the disaster response safety center belonged to the emergency operations facility (EOF), but the radiation emergency plan did not mention it at all. The main tasks of disaster response safety center were the movement, arrangement and connection of mobile emergency response facilities, on-site construction of other emergency response facilities, and on-site road restoration. According to the accident management plan, the movement, arrangement, and connection of mobile facilities (i.e., mobile generators, mobile pumps, multi-purpose communication relay facilities), which were considered very important for the prevention and mitigation of serious accidents, were under the supervision of the disaster response safety center. It was stipulated that the operation was carried out with the cooperation of a regular emergency organization, and that the start, operation and stop of mobile equipments were to be performed under the supervision of the emergency operation team supported by the regular emergency organization. Since this organization structure and assignment of duties could not be confirmed in radiation emergency plan, it was necessary to revise and improve the radiation emergency plan for the successful operation of mobile equipments and to link them with the accident management plan.

Acknowledgements

This research was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS) using the financial resources granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 1805018). This research was also supported by the 2022 Research Fund of the KEPCO International Nuclear Graduate School (KINGS), the Republic of Korea.

Keywords: Accident Management Plan, Radiation Emergency Plan, Emergency Response Facility, Mobile Equipment

Measurement of Alpha Radiation Dose in Uranium-using Facility

Seonho Noh*, Youngwoo Kwon, and Youngsang Lee

Korea Atomic Energy Research Institute (KAERI)

*nohsh@kaeri.re.kr

The Korea Atomic Energy Research Institute operates the Nuclear Cycle Experimental Research Facility which has radiation controlled area in the laboratory with the aim of realizing pyroprocessing technology.

In this Facility, depleted Uranium feed material and a depleted Uranium mixed with some surrogate material are used for performing experiments. Therefore the facility is using uranium, users should be careful of radiation. This paper will explain the radiation protection of the Nuclear Cycle Experimental Research facility and will also explain how much alpha radiation comes out from the facility. The RMS (Radiation Monitoring System) detector is made by CANBERRA and the model name is ICAM. ICAM RMS is the detector which can detect Alpha Radiation by absorbing the air in the facility. The RMS detector is installed in the HVAC room on the third floor to check the air contamination through the chimney. The RMS is connected to the air ventilation line for detecting Alpha radiation in the whole facility. Experiment was performed for two weeks to check the radiation level and the air ventilation fan continued to operate 24 hours a day. the results are below the required value which is 0.1 Bq/m^3 , indicating that the facility is safe in terms of radiation safety management.

Keywords: Pyroprocessing Facility, RMS, Radiation Monitoring System

Simulations on Routine Emission of Radioxenon in the Northern Hemisphere Using LADAS Model: Case Study in 2014

Kihyun Park*, Byung-Il Min, Sora Kim, Jiyeon Kim, and Kyung-Suk Suh

Korea Atomic Energy Research Institute (KAERI)

*khpark77@kaeri.re.kr

As part of the third ATM Challenge, we performed a series of atmospheric dispersion simulations for routine releases of Xe-133 from ordinarily operating nuclear facilities such as Medical Isotope Production Facilities (MIPFs), Nuclear Power Plants (NPPs), and Research Reactors (RRs) in the Northern Hemisphere using our ATM, Lagrangian Atmospheric Dose Assessment System (LADAS), with Numerical Weather Prediction (NWP) data produced by the Korea Meteorological Administration (KMA). The simulation time period is 6 months, from June to November in 2014, and we used the stack emission data except for CNL (Canada) and IRE (Belgium) in accordance with the scenario of the third ATM Challenge 2019. In addition, the simulations were done individually for all MIPFs, NPPs, and RRs. We utilized 3-hourly KMA's Unified Model Global Data Assimilation and Prediction System (UM-GDAPS) data with $0.35^{\circ} \times 0.23^{\circ}$ horizontal resolution as input meteorological fields and extracted hourly time series results for Xe-133 activity concentrations with few different resolutions such as $0.5^{\circ} \times 0.5^{\circ}$, $0.35^{\circ} \times 0.23^{\circ}$, and $0.1^{\circ} \times 0.1^{\circ}$ at several IMS stations in the Northern Hemisphere which were in normal operation in 2014. Considering previously reported values of daily Xe-133 release amounts for CNL and IRE, measured signals at some IMS stations (such as CAX17, DEX33, SEX63, and USX75) were well reproduced from the simulation results.

Keywords: ATM, NWP, KMA, UM-GDAPS, LADAS, IMS

A Study on the Evaluation of the Moving Path and Expected Annual Exposure Dose of Workers Considering the Work Zone in Movable Radionuclide Analysis System

Sang Heon Lee, Min Ho Lee, and Jong Soon Song*

Chosun University

*jssong@chosun.ac.kr

In Korea, the decommissioning of nuclear power plants is being prepared, and a large amount of radioactive waste is expected to be generated. In particular, clearance level waste, which accounts for more than 90%, requires prevention of cross-contamination and prompt classification. In this study, the possible exposure route and the derivation of exposure dose for worker exposure management in a movable analysis system that can be analyzed onsite were studied.

The movable radionuclide analysis system is divided into a preparatory room, a sample storage room, a radioanalysis room, a laboratory, and a waste storage room. It consists of one radioanalysis worker and one pre-treatment worker, and the main radiation exposure is expected to occur in the movement path in the sample storage room, radioanalysis room, and laboratory.

The source term for the exposure evaluation, the annual usage dose presented in the radiation safety report in the movable radionuclide analysis system was used. The input data for the evaluation of the external exposure dose under normal circumstances (exposure situation, working hours, distance, etc.) is referenced at facility specifications. The internal exposure dose evaluation was assumed to be acute exposure (1 hour) assumed as internal pollution due to the drop in liquid sample during the pre-treatment work.

As an evaluation method, a method using a calculation formula and a method using an evaluation code was performed. For the evaluation of exposure dose using the calculation formula, a preliminary evaluation was performed using the point source method, the point kernel method, and intake and dose conversion factors. In addition, VISIPLAN and IMBA codes were used to evaluate exposure dose using the evaluation code, and the input data were supplemented for evaluation.

As a result of the evaluation, the annual exposure dose limit of 20 mSv was satisfied for both normal and non-normal situations. In future research, it is planned to derive the evaluation results by particular scenarios for the detailed movement route and evaluation time according to the work process in the mobile radionuclide analysis.

Acknowledgements

This work was supported by the Nuclear Power Core Technology Development Program of the Korea Institute of Energy Technology Evaluation and Planning (KETEP), which was granted financial resources from the Ministry of Trade, Industry & Energy, Republic of Korea (No. 20191510301240).

Keywords: Movable Radionuclide Analysis System, Decommissioning Waste, Exposure Dose, Exposure Pathway

Evaluation of Surface Contamination Measurement Results Based on the Recommendations of the International Organization for Standardization

Chaehee Kim* and Insu Chang

Korea Atomic Energy Research Institute (KAERI)

*chay@kaeri.re.kr

Surface contaminants may attach to surfaces or objects in the radiation controlled area to cause radiation exposure, or spread out to the general environment by person and object exiting the radiation workplace. Accordingly, in radiological safety control, surface contamination monitoring is one of the important factors in workplace monitoring. When obtaining the measurement results for the monitoring, the results are accompanied by uncertainty since measurements contain numerous errors. Accordingly, the International Organization for Standardization (ISO) has published the ISO 7503 series which is comprehensive and detailed guidelines on the measurement and evaluation of surface contamination. ISO 7503-3 especially presents a mathematical model for the contamination measurement and provide calculation guidelines on measurement uncertainty evaluation, decision threshold and detection limit.

This paper is focused on reevaluating and comparing the surface contamination monitoring method applied to radiation safety management practice and its results based on the measurement and evaluation method set by the International Organization for Standardization. The evaluation was performed in accordance with ISO 7503, and the current reporting method for measurement results was compared with the method recommended in ISO 11929 publication.

Acknowledgements

This work was supported by the Korea Atomic Energy Research Institute (KAERI) (NTIS No. 1711173789).

Keywords: Surface Contamination, Decision Threshold, Detection Limit, ISO 7503, ISO 11929

Assessment of Dose Rate and Radioactivity Ratio in Some Areas of Jeju Island

Byungchae Lee*, Jihyun Yu, Jonghoa Kim, and Jangsoo Suh

Sae-An Eneritech Corp.

*bcleee@sae-tec.com

In this research, the dose rate was measured using backpack-type scan survey device at 4 sites on Jeju Island, and the radioactivity ratio for each nuclide was evaluated using an high-purity germanium (HPGe) detector. As a result of measuring the dose rate with a backpack-type scan survey device, the average dose rate was the lowest in the measurement site 3 at 0.049 $\mu\text{Sv/h}$, and the highest in the measurement site 1 at 0.066 $\mu\text{Sv/h}$. The average dose rate of the 4 sites on Jeju Island was 0.056 $\mu\text{Sv/h}$, and the dose rate on Jeju Island was lower than that of other regions. The data acquired by scan survey were interpreted using classed post and gridding function of surfer program. The radioactivity ratio of each nuclide in the gamma spectrum measured by the HPGe detector was the highest for K-40 with an average of 87.62%, and the lowest for Pb-214 with an average of 0.63%. In the case of the Jeju Island site, Cs-137 was detected, and the average radioactivity ratio of Cs-137 was 3.27%, which was the background level. The results of this research can be used as basic data on the radioactivity ratio for each nuclide and dose rate at the Jeju Island site. Further studies on the assessment of dose rates and radioactivity ratios in other regions are needed.

Keywords: Jeju Island Site, Scan Survey, Dose Rate, Radioactivity Ratio

Analysis of Risk Assessment Methodologies in Radioactive Waste Overland Transportation Risk Assessment Codes for Transportation Accident

Min Woo Kwak, Hyeok Jae Kim, Seong Yeon Lee, and Kwang Pyo Kim*

Kyung Hee University

*kpkim@khu.ac.kr

Currently, low and intermediate-level radioactive wastes and spent nuclear fuels are continuously generated in Korea. For the disposal of the radioactive wastes, the transport demand is expected to increase. Prior to transportation, it is necessary to evaluate the radiation risk of transportation to confirm that is not high. In Korea, there is no transportation risk assessment code that reflects domestic characteristics. Therefore, foreign assessment codes are used. In this study, before developing the overland transportation risk assessment code that reflects domestic characteristics, we analyzed the radiation risk assessment methodology in transportation accident codes developed in other countries.

RADTRAN and RISKIND codes were selected as representative overland transportation risk assessment codes. For the two codes we analyzed accident scenarios, exposure pathways, and atmospheric diffusion. In RADTRAN, the user classifies accident severity for possible accident scenarios, and the user inputs the probability for each accident severity. On the other hand, in the case of RISKIND, the accident scenarios are classified and the probabilities are determined according to the NRC modal study (LLNL, 1987) in consideration of the cask impact velocity, cask impact angle, and fire temperature. In the case of RISKIND, the accident scenarios are applied only to transportation of spent nuclear fuel, and cannot be defined for low and intermediate-level radioactive waste. However, in the case of RADTRAN, since the severity and probability of accidents are defined by user, it can be applied to low and intermediate-level radioactive wastes.

As the exposure pathways considered in transportation accident, both RADTRAN and RISKIND consider external exposure (cloudshine and groundshine), and internal exposure (inhalation, resuspension inhalation and ingestion). In the case of RADTRAN, additionally, external exposure due to loss of shielding (LOS) is considered.

Atmospheric diffusion calculation is essential to determine the extent to which radioactive materials are diffused. In both RADTRAN and RISKIND, atmospheric diffusion calculations are based on Gaussian diffusion model. Users must input Pasquill stability class, release height, heat release, wind speed, temperature and mixing height, etc. Additionally, RADTRAN can input weather information relatively simply by inputting only the Pasquill stability class fraction and selecting the US average weather option. This study results will be used as a basis for developing radioactive waste overland transportation risk assessment code that reflects domestic characteristics.

Acknowledgements

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea (No. 2106041).

Keywords: Radiation Risk Assessment, Radioactive Waste, Transportation Accident, RADTRAN, RISKIND

Discussion on Calculation Procedure of Kr-85 Concentration in Gaseous Effluent and Emission Evaluation

Suhyun Choi*, Cheolhyun Kang, Jongil Lee, and Insu Chang

Korea Atomic Energy Research Institute (KAERI)

*choish@kaeri.re.kr

Kr-85 has a half-life of 10.7 years and it stays in the atmosphere for a long time. However it does not accumulate as an noble gas but only emits beta particles. Therefore its contribution to environmental radiation dose is lower than any other radionuclides. Kr-85 is one of the main fission products produced by nuclear fission reaction and artificial radionuclide that does not exist in nature. For these reasons, monitoring Kr-85 from the atmosphere is meaningful so that the nuclear-related facilities are recommended to control and regulate environmental emissions.

Post Irradiation Examination Facility (PIEF) which located in KAERI is a facility that conducts various material and chemical experiments using the irradiated nuclear fuels. Therefore, various radionuclides can present in gaseous effluent including Kr-85.

To prevent the environmental hazards and guarantee the radiation safety of the public, nuclear facilities are recommended to be equipped with stack radiation/radioactivity monitoring system, so that the Kr-85 concentration in gaseous effluent is controlled within the regulatory criteria.

Particularly, the Kr-85 concentration of gaseous effluent is commonly monitored by the stack monitoring system connected to the process ventilation system from the hot cell. The monitoring system supply the information such as beta count rate, dose rate and flow rate, etc. Due to the concentration of Kr-85 in gaseous effluent is subject to regulatory guide lines, a systemized procedure for calculating Kr-85 concentration of the stack exhaust is necessary. Furthermore, the emission should be monitored whether it satisfies the regulatory standard or does not.

This paper performed discussion on the process of calculating the concentration of Kr-85 in the gaseous effluent of PIEF stack from the monitoring system (NGM209, MGP), and the amount of Kr-85 over the last 2 years emissions was calculated. In addition to calculating effluent rate of radioactive Kr-85, the Minimum Detectable Concentration (MDC) and Decision Threshold (SD) were calculated. As a result, the calculated Kr-85 concentration was below the SD during the entire period. It is considered that there are no environmental emissions of Kr-85.

Acknowledgements

This work was supported by the Korea Atomic Energy Research Institute (KAERI) (NTIS No.171 1173789).

Keywords: Stack Monitoring, Kr-85, Gaseous Effluent, Minimum Detectable Concentration (MDC), Decision Threshold (SD)

7분과

방사화학(Oral)

A Review of Chemical Speciation and Thermodynamic Studies of Plutonium in KAERI

Hye-Ryun Cho*

Korea Atomic Energy Research Institute (KAERI)

*hrcho@kaeri.re.kr

For the safety assessment of the high-level radioactive waste (HLRW) disposal, the thermodynamic data such as solubility products, formation constants of complexes, redox equilibrium constants of radionuclides, and their reaction enthalpy and entropy are required. In order to recommend and summarize the reliable data, thermodynamic databases (TDB) have been persistently developed through the OECD-NEA TDB projects and an updated TDB of actinides has been recently published in 2020. To date, reliable data for Pu reactions are scarce due to the possibility of coexistence of four different oxidation states, Pu(III-VI) by redox equilibria in solutions. To determine the thermodynamic data for the reaction of each Pu oxidation state, it is necessary to precisely control the oxidation state and quantitatively analyze all reactants, products and bi-products by using highly sensitive speciation techniques. Since 2004, the nuclear chemistry research team in KAERI has been focused on developing techniques for the sensitive chemical speciation by using laser-based spectroscopy and determining thermodynamic data of actinides such as U, Pu, Am.

In this paper, chemical speciation and thermodynamic studies on Pu in KAERI are reviewed. A combination of a commercial spectrophotometer and a capillary cell was adopted for a sensitive chemical speciation of Pu(III-VI) in solutions. A sensitive detection of trace amount of Pu colloids was carried out with the laser-induced breakdown detection (LIBD) system. Pu(VI) complexation with hydroxide or carbonate ions were investigated under strong oxidation conditions controlled with hypochlorite (NaOCl). The solubility product of $\text{Pu}(\text{OH})_3(\text{am})$ and formation constant of $\text{Pu}(\text{III})\text{-OH}$ species were determined by a combination of wet-chemistry experiments and several analysis methods of spectrophotometry, LIBD, radiometry under a strong reducing condition controlled by electrochemistry. More recently, we reported the reaction enthalpy and entropy data for the formation of $\text{Pu}(\text{OH})^{2+}$ and the dissolution of $\text{Pu}(\text{OH})_3(\text{am})$. A preliminary data for reaction between Pu(III) and organic matter will be presented.

Keywords: Plutonium, Thermodynamic Data, Chemical Speciation, Safety Assessment, HLRW Disposal

Chemical Behavior of Niobium(V) in Cementitious Conditions: Solubility and Sorption

Yongheum Jo^{1,2,*}, Nese Cevirim-Papaioannou², Krassimir Garbev², Markus Fuss², Karsten Franke³,

Oliver Dieste Blanco², Benny de Blochouse⁴, Marcus Altmaier², and Xavier Gaona²

¹Korea Atomic Energy Research Institute (KAERI)

²Karlsruhe Institute of Technology

³Helmholtz-Zentrum Dresden-Rossendorf

⁴ONDRAF/NIRAS

*yhjo@keari.re.kr

Niobium (Nb) is present in Ni-based alloys and stainless steels used in nuclear reactors as structural materials. Nb-93 is a naturally occurring and stable isotope of niobium and Nb-94 (half-life = 20,000 years) is produced by neutron activation of Nb-93. Nb-94 can be present in waste streams from dismantling of nuclear power plants and treatment of the primary coolant circuit. Hence, the radioactive wastes containing active Nb-94 are disposed of in the repositories for low- and intermediate-level waste (LILW). Nb predominantly exhibits a pentavalent oxidation state (i.e., +V) within the stability field of water. Cementitious materials (concrete, mortar, and grout) are extensively utilized in LILW disposal systems as structural components and chemical agents for the stabilization of waste. Solubility defines the source term (i.e., upper concentration limit) in the repository system. However, the solubility behavior of Nb in cementitious systems at high pH remains ill-defined, and information available on the Nb solid phases controlling the solubility is scarce and often ambiguous. Sorption on cement-based materials is one of the main mechanisms controlling the retention of niobium(V) in a LILW repository, and distribution coefficients (R_d) are necessary to evaluate the retention capacity by sorption in the safety assessment of disposal systems. Available sorption data of Nb(V) on cement showed a large discrepancy in R_d , moreover, no sorption data is available for Nb(V) under conditions characterizing the first degradation stage of cement (young cement condition) at pH 13 – 13.5.

In this context, the solubility of Nb was extensively investigated in porewater conditions representative of the cement degradation stage I, as well as in CaCl_2 - $\text{Ca}(\text{OH})_2$ systems. Special focus was given to the accurate characterization of the solubility-controlling solid niobium phases. We also studied the sorption of Nb(V) by hardened cement pastes (HCP) and calcium silicate hydrates (CSH, major hydrate of HCP). This work provides the results on R_d , sorption isotherm and sorption mechanisms of Nb(V). Besides, the impact of ISA (polyhydroxycarboxylic acid generated by the degradation of cellulose) on Nb(V) sorption and the dissolution of cement materials was investigated.

Keywords: Niobium, Solubility, Sorption, Cementitious Systems

Application of Machine Learning to Predict the Chemical Reactions at Solid-Water Interfaces

Jun-Yeop Lee* and Do-Hyun Kim

Pusan National University (PNU)

*jylee@pusan.ac.kr

The sorption/adsorption behavior of radionuclides, usually occurring at the solid-water interface, is considered to be one of the primary reactions that can hinder the migration of radiotoxic elements contained in the spent nuclear fuel. In general, various physicochemical properties such as surface area, cation exchange capacity, type of radionuclides, solid-to-liquid ratio, aqueous concentration, etc. are known to provide a significant influence on the sorption/adsorption characteristics of target radionuclides onto the mineral surfaces. Therefore, the distribution coefficient, K_d , inherently shows a condition-dependent behavior according to those highly complicated chemical reactions at the solid-water interfaces. Even though a comprehensive understanding of the sorption behavior of radionuclides is significantly required for reliable safety assessment modeling, the number of the chemical thermodynamic model that can precisely predict the sorption/adsorption behavior of radionuclides is very limited.

The machine-learning based approaches such as random forest, artificial neural networks, etc. provide an alternative way to understand and estimate complicated chemical reactions under arbitrarily given conditions. In this respect, the objective of this study is to predict the sorption characteristics of various radionuclides onto major bentonite minerals, as backfill materials for the HLW repository, in terms of the distribution coefficient by using a machine-learning based computational approach. As a background dataset, the sorption database previously established by the JAEA was employed for random forest machine learning calculation. Moreover, the hyperparameters such as the number of decision trees, the number of variables to divide each node, and random seed numbers were controlled to assess the coefficient of determination, R^2 , and the final calculation result. The result obtained in this study indicates that the distribution coefficients of various radionuclides onto bentonite minerals can be reliably predicted by using the machine learning model and sorption database.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (No. NRF-2021M2E1A108520412).

Keywords: Machine Learning, Sorption, Distribution Coefficient, JAEA-SDB

Experimental Analysis of HTO Adsorption on Activated Carbon in Tritium-Contaminated Water

Jei-Won Yeon* and Ji Eun Yang

Korea Atomic Energy Research Institute (KAERI)

*yeonysy@kaeri.re.kr

Though many treatment technologies of contaminated water have been developed for a long time, it is still difficult to find a suitable method for large volumes of low radioactivity tritium-contaminated water. For this reason, most of the tritium-contaminated water been discharged to the biosphere or been stored in a special control area as radioactive waste. Activated carbon is a common material, but since there are few data on the treatment of tritium-contaminated water, its adsorption behavior to HTO is worth studied. In our study, for the tritium-contaminated water having a low radioactivity concentration (350-480 Bq/g), adsorption experiments were performed with activated carbon. The effects on the selective adsorption of HTO were investigated for temperature (5-55°C), hydrogen peroxide (1-10wt%) and activated carbon reuse (1-6 times) under non-equilibrium conditions. The treatment of activated carbon significantly reduced the radioactivity of tritium-contaminated water around 60 minutes of adsorption time. In order to clearly analyze the experimental results, positive factors and negative factors on the HTO selectivity were separately evaluated according to the adsorption time. Temperature and the reuse of activated carbon were evaluated as negative factors for HTO selectivity of activated carbon, whereas hydrogen peroxide (> 5wt%) was evaluated as a positive factor. By the evaluation method of separating the influencing factors into two types, the adsorption experimental results of HTO could be understood more clearly.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT: the Ministry of Science and ICT) (2021M2E3A3040092, RS-2022-00144202, 2017M2A8A4015281). Additionally, this work was partially supported by the KAERI Institutional Program (Project No. 522330-22).

Keywords: Adsorption, HTO, Activated Carbon, Tritium-contaminated Water

Radiochemical Separation of Nb Isotopes in Radioactive Waste Samples From the Nuclear Power Plant Using Anion Exchange Resin

Myung Ho Lee*, Byoung Chul Song, Kyung Sik Kim, Young Jae Maeng, Choon Sung Yoo,

and Byoung Chul Kim

Korea Reactor Integrity Surveillance Technology (KRIST)

*myholee@krist.co.kr

This study presents a rapid and quantitative radiochemical separation method for Nb isotopes in radioactive waste samples from the nuclear power plant with anion exchange resin after Fe coprecipitation. After radionuclides were leached from the radioactive waste samples with concentrated HCl and HNO₃, the Nb isotopes were coprecipitated with Fe after filtering the leaching solution with 0.45 micron HA filter, while the Sr, Tc and Ni isotopes were in the solution. The Nb isotopes were separated in HCl medium with anion exchange resin. The purified Nb isotopes were measured using a low level liquid scintillation counter after installing quenching curve with standard Nb-94 isotopes. The separation method for Nb isotopes investigated in this study was applied to neutron dosimeter samples from the nuclear power plant after validating the Nb activity concentration with gamma spectrometry system.

Keywords: Radiochemical Separation Method, Nb Isotopes, Liquid Scintillation Counter, Radioactive Waste samples, Gamma Spectrometry System

Modified Separation Method for ^{99}Tc , ^{94}Nb , ^{90}Sr , ^{55}Fe , $^{59/63}\text{Ni}$, ^{60}Co , ^{137}Cs

Heung Nae Lee*, Kwy Hwan Cho, Kyu Kwang Kang, Hyeong Min Park, Jun Hwi Bang, and Yun Han Lee

Korea Standards Analysis Co. LTD.

*heungnae.lee@ko-sa.co.kr

According to the Nuclear Safety and Security Commission (NSSC) Notice No. 2021-26 “Delivery Regulations for the Low- and Intermediate Level Radioactive Waste (LILW)”, the activity of ^3H , ^{14}C , ^{55}Fe , ^{58}Co , ^{60}Co , ^{59}Ni , ^{63}Ni , ^{90}Sr , ^{94}Nb , ^{99}Tc , ^{129}I , ^{137}Cs , ^{144}Ce , and gross alpha must be identified. Currently, the scaling factor of the dry active waste (DAW) for LILW is applied as an indirect evaluation method in Korea. The analyses are used the destructive methods and ^{55}Fe , ^{60}Co , ^{59}Ni , ^{63}Ni , ^{90}Sr , ^{94}Nb , ^{99}Tc , and ^{137}Cs , which are classified as nonvolatile nuclides, are separated through sequential separation and then measured by gamma detector, liquid scintillation counter (LSC), alpha/beta total counter (Gas Proportional Counter, GPC), and ICP-MS. We will introduce how to apply the existing nuclide separation method and improve the measurement method to supplement it.

Acknowledgements

Periodic Verification (5th) Service of the Scaling Factor for Radioactive Waste.

Keywords: LILW, DAW, Scaling Factor, Non-volatile Nuclides

Analysis of Light Rare-Earth Elements Using Gas-Pressurized Extraction Chromatography

Jihye Kim*, Namuk Kim, and Sang-Eun Bae

Korea Atomic Energy Research Institute (KAERI)

*jihyekim21@kaeri.re.kr

A new method for chemical separation of light rare-earth elements (LREEs) using gas-pressurized extraction chromatography (GPEC) is described. GPEC is a microscale column chromatography system that features a constant flow of solvents (0.1 mL/min), which is created by pressurized nitrogen gas. The separation column with a Teflon tubing was packed with LN resin. We evaluated the separation of Ba, La, Ce, and Nd using various elution solvents. Here, we applied the natural isotopes of LREEs (La-139, Ce-140, and Nd-144) and barium (Ba-138) instead of radioactive isotopes for the preliminary test and reducing unnecessary radioactive waste. The column reproducibility of the proposed GPEC system ranged from 2.4% to 4.9% with RSDs of recoveries, and the column-to-column reproducibility ranged from 3.1% to 6.3% with RSDs of recoveries. This proposed GPEC method provides robust analysis and facilitates production of lesser chemical wastes and faster separation owing to the use of low solvent volume compared to traditional column chromatography.

Keywords: Light Rare-Earth Elements, Separation, Extraction Chromatography, Lanthanum, Cerium, Neodymium

7분과

방사화학(Poster)

Validation Result of Combustion Method for Simultaneous Analysis of ^3H and ^{14}C

Kyungwon Suh^{1,*}, Kwang Eun Lee¹, Jung Bo Yoo¹, and Tae-Hong Park^{1,2}

¹Korea Atomic Energy Research Institute (KAERI)

²University of Science and Technology (UST)

*kyungwon0707@kaeri.re.kr

Combustion method has been widely used in the analysis of ^3H and ^{14}C in various types of radioactive wastes since X. Hou reported the analysis of ^3H and ^{14}C in graphite and concrete for decommissioning of nuclear reactor. In this work, it was demonstrated that the validation result of combustion method for the simultaneous analysis of ^3H and ^{14}C in non-combustible radioactive wastes. To validate the combustion method, ^3H and ^{14}C spiked to sea sand and soil were prepared and each simulated sample was combusted with the accordance to a combustion temperature program. The validation of combustion method was assessed by the radioactivity recovery, radioactivity deviation, and relative standard deviation of measured radioactivity. The results of radioactivity recovery, radioactivity deviation, and relative standard deviation of ^{14}C were 100~105%, less than 7%, less than 5%, respectively. In addition, ^3H showed about 90% of radioactivity recovery, less than 20% of radioactivity deviation, and less than 5% of relative standard deviation. It will be provided that the validation results of combustion method in detail.

Keywords: Combustion Method, Simultaneous Analysis, Tritium, C-14

Synthesis and Characterization of Ca-Silicate Bearing Uranium Minerals

Wansik Cha*, Junghwan Park, Hye Ran Noh, Euo Chang Jung, Sang Ho Lim, Hye-Ryun Cho,

and Hee-Kyung Kim

Korea Atomic Energy Research Institute (KAERI)

*wscha@kaeri.re.kr

Nuclear spent fuel (SNF) disposal in deep geological repositories is considered as one of sound options for the long-term and safe sequestration of radiotoxic SNF and the sustainable use of nuclear energy. The chemical behaviors of various radionuclides originated from SNF should be well understood to evaluate the migrational behaviors of radionuclides and their reactions and interactions with various geochemical components. Formation of secondary minerals, colloids, other insoluble precipitates is of interest since the concentrations of radionuclides in groundwaters can be limited by the solubility of those solid phases. Particularly when evaluating their solubility, the use of well-defined solid materials in terms of chemical composition and molecular structure is crucial to obtain reliable measurement results. In this study, a synthetic calcium uranyl silicate (Ca-U(VI)-silicate, or uranophane) was prepared and characterized by using various analytical methods including powder X-ray diffraction (pXRD), scanning electron microscopy/energy dispersive X-ray spectrometry (SEM/EDX), and vibrational (FTIR and Raman) spectroscopies.

Uranyl silicate minerals are significant to the disposal of nuclear wastes. Our simulation demonstrates that uranophane ($\text{Ca}[\text{UO}_2\text{SiO}_3\text{OH}]_2 \cdot 5\text{H}_2\text{O}$), one having a U:Si ratio of 1:1, can be a mineral species limiting U(VI) solubility under groundwater conditions in Korea. For the preparation of Ca-U(VI)-silicate, we applied a two-step hydrothermal synthetic procedure reported in literature with modification. Briefly, we conclude that the obtained mineral phase is the ‘ α -uranophane’; our characterization results show that the structural and spectroscopic properties of the synthetic Ca-U(VI)-silicate agree well with those of α -uranophane. For instance, the pXRD patterns obtained from the solid show nearly identical diffraction peak positions with those from the reference XRD pattern. From IR and Raman spectroscopy it is noticed that the stretching modes of UO_2^{2+} and SiO_4^{4-} ions result in strong absorption bands in a region of $700 \sim 1,100 \text{ cm}^{-1}$. Elemental compositions of the synthetic solids were also estimated by using EDX analysis, which results in a Ca:U:Si ratio close to 1:2:2 on average. However, we found that it is difficult to obtain good crystallinity of uranophane, which can be observable by using SEM and its image analysis. We believe that this work serves as a model study to provide synthetic routes of radionuclide-related mineral phases and applicable solid phase characterization methods. In the presentation, the potential use of the U(VI)-silicate solid phase for the upcoming groundwater solubility measurements will be discussed.

Keywords: Hexavalent Uranium, Silicate Mineral, Uranophane, X-ray Diffraction, Solubility, FTIR/Raman Spectroscopy

Influence of Effective Thermal Conductivity on the Radiolytic and Chemical Environment Within the CRUD Deposits

Seungjin Seo¹, Hwajeong Han², Byunggi Park², and Sungyeol Choi^{1,*}

¹Seoul National University (SNU)

²Soonchunhyang University

*choisys7@snu.ac.kr

Corrosion products generated from the oxidation of structure materials are deposited on the surface of coolant systems, forming CRUD (Corrosion Related Unidentified Deposits). The CRUD deposition on the fuel surface has influenced the heat transfer through the fuel rod. When CRUD was deposited on a fuel surface, heat resistance may increase, and this increase in heat resistance leads to the increase in temperature distribution from cladding to coolant. Also, the temperature distribution is related to the radiolytic and chemical reactions within the CRUD deposits. This influence may be enough to change the pH distribution within the CRUD deposits. To estimate the influence of thermal resistance, the composition, microstructure, and vapor fraction within the CRUD should be considered, by investigating the thermal conductivity model of CRUD deposits. Therefore, in this study, the CRUD thermal conductivity was studied through the literature study, by considering composition, capillary flow characteristics, and vapor fraction. For the uncertainty parameters, a sensitivity study was conducted to check the degree of influence on thermal conductivity. The effective thermal conductivity was applied to the radiochemistry model within the CRUD deposits and an analysis of the influence in radiolysis reaction within the CRUD deposits with a fixed thickness.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (No. RS-2022-00144515).

Keywords: CRUD, Effective Thermal Conductivity, Radiolysis, Chemistry, Capillary Flow, Vaporization

Thermal NaCl-MgCl₂ Salt Purification for Reducing Reactor Vessel Corrosion

Wonseok Lee¹, Wonseok Yang², Jihun Kim¹, Hyeongbin Kim¹, and Sungyeol Choi^{1,*}

¹Seoul National University (SNU)

²Korea Advanced Institute of Science and Technology (KAIST)

*chosys7@snu.ac.kr

Molten chloride salts are promising candidates as a coolant for Molten Salt Reactors (MSRs) because of their low cost, high specific heat transfer, and thermal energy storage capacity. The NaCl-MgCl₂ eutectic salts have enormous latent heat (430 kJ/kg) and financial advantage over other types of molten chloride salt. Despite the promise of the NaCl-MgCl₂ eutectic salt, problems associated with structural material corrosion in the MSR system remain. The hygroscopicity of NaCl-MgCl₂ and high MSRs operating temperature accelerate corrosion within structural alloys. Especially, MgCl₂ reacts with H₂O in the eutectic salt to produce HCl and Cl₂, which are known to further exacerbate corrosion by the chlorination of structural materials. Therefore, several studies have worked to purify impurities associated with MgCl₂, such as H₂O. Thermal salt purification of NaCl-MgCl₂ eutectic salt is one method that reduces HCl and Cl₂ gas generation. However, MgO and MgOHCl are generated as the byproduct of thermal purification through a reaction between MgCl₂ and H₂O. The corrosion behavior of MgO within structural alloys after thermal treatment is not well known.

This paper demonstrates corrosion behavior within structural alloy after thermal treatment at various temperature profiles of the NaCl-MgCl₂ eutectic salt. According to the temperature range, MgCl₂·H₂O are separated at 100~200°C, and MgOHCl and HCl begin to occur at 240°C or higher. Finally, MgOHCl produces MgO and HCl at 500°C or higher temperatures. After thermal treatments, the H₂O, MgOHCl, and MgO content were measured by Thermo Gravimetric Analyzer (TGA) to evaluate significant products causing corrosion. The structural materials were analyzed by the Scanning Electron Microscope-Energy Dispersive Spectroscopy (SEM-EDS) and using the mass change method to observe the type of localized corrosion, the corrosion rate, and the corrosion layer thickness.

This study is possible in that it can reduce economic costs by reducing the essential use of expensive, high-purity molten salts because it is related to a substantial financial cost problem considering the amount of molten salt used in industrial sites.

Acknowledgements

This work was supported through the National Research Foundation of Korea (NRF), funded by the Ministry of Science and ICT directly (grant number: NRF-2022M2D4A1052797).

Keywords: Molten Salt Reactor, Heat Treatment, Molten Salt Purification, Impurity Removal, Corrosion Resistance

Magnesium Chloride Dehydration Methods Using Thermal and Electrochemical Treatments

Jihun Kim¹, Wonseok Yang², Hyeongbin Kim¹, Wonseok Lee¹, Taeho Jang¹, and Sungyeol Choi^{1,*}

¹Seoul National University (SNU)

²Korea Advanced Institute of Science and Technology (KAIST)

*choisys7@snu.ac.kr

Molten salts based on magnesium chloride can be used in the nuclear power reactor because they have a high heat capacity and heat stability, and allow for a faster neutron spectrum. However, magnesium chloride is highly hygroscopic, leading to the formation of moisture-related impurities, which result in the corrosion of structural materials and negatively affect the operation of the reactor.

The dehydration of magnesium chloride is studied using both thermal and electrochemical treatments. According to previous studies, water impurities in magnesium chloride molten salt transform into magnesium oxide over 650 degrees Celsius. The temperature profile of the molten salt is suggested to separate magnesium chloride and magnesium oxide, focusing on cooling rate near the freezing point of magnesium chloride. Two layers separated by a phase boundary on the salt surface appear due to the density difference between magnesium oxide and magnesium chloride. Further, the removal of oxide ions remaining in the molten salt is carried out by electrochemical treatment. Two different cells, each consisting of two electrodes, are used. One cell is composed of graphite anode and nickel cathode. The other is composed of tin oxide anode and nickel cathode. As the reaction proceeds, carbon dioxide and oxygen are generated in graphite and tin oxide, respectively, and magnesium electrodeposition occurs at the cathode. The amount of purified magnesium oxide is measured to the endpoint, which is notified by the reduced current. The efficiency of each method is compared by measuring the weight ratio of the purified part to the unpurified part. Thermogravimetric analyzer (TGA) and UV-vis spectroscopy are used to check the quality of the purified part. Only magnesium oxide remains at a temperature above the boiling point of magnesium chloride. Therefore, the amount of magnesium oxide in the purified part can be measured by the mass change of the salt through the TGA method. For UV-vis spectroscopy, the transmittance is measured which depends on the weight percent of the impurities in the purified part.

The suggested purification method using both thermal and electrochemical treatment is assessed quantitatively and qualitatively. It is expected that hygroscopic molten salts other than magnesium chloride will be able to be dehydrated through the above process.

Acknowledgements

This work was supported through the National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT directly (grant number: NRF-2022M2D4A1052797).

Keywords: Molten Salts, Magnesium Oxide, Temperature Profile, Electrodeposition, Salt Purification

Corrosion Failure Analysis of the Welded Part in the Multi-purpose Molten Salt Experiments

Hwa Jeong Han, Hyunjin Boo, Jonghyeon Kim, and Byung Gi Park*

Soonchunhyang University

*byunggi@sch.ac.kr

In this study, molten salt experiments were performed using a multi-purpose molten salt experimental loop to evaluate the corrosion and thermodynamic properties of the molten salt. The multi-purpose molten salt experimental loop is made of 1-inch austenitic 316 stainless steel, and 1/4-inch austenitic 316 stainless steel tubes were welded on the surface of a 1-inch pipe to measure temperatures of molten salt. During the experiment, the molten salt leaked due to corrosion of the welded part of the 1/4-inch tube connected to the 1-inch pipe. Therefore, the cause of corrosion damage of the leaked welded part was analyzed. The effect of NaCl-MgCl₂ salt selected as the molten salt on corrosion failure was considered. And based on the operation data of molten salt experiments, the time of occurrence of the issue was estimated. Lastly, the cause of corrosion failure was estimated by comparing and analyzing the pipe shape before and after failure using SEM-EDS.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP: Ministry of Science, ICT and Future Planning) (No. 2020M2D4A1070724).

Keywords: Molten Salt Reactor, Molten Salt, Liquid fuel, Thermal Convection Loop

Study on Ion Exchange Resin for Separation/Purification of Radioactive Strontium

Yeong Su Ha*

Osong Medical Innovation Foundation

*gksksrkd@naver.com

Cardiovascular disease remains as one of the most common causes of high morbidity and mortality worldwide, despite remarkable medical advances in recent decades. Non-invasive techniques play a preeminent role in prevention of cardiovascular disease by diagnosing it at an early stage and guiding optimal patient management. Nuclear imaging is one of the most powerful means available for non-invasive diagnosis and management of poorly perfused myocardial region resulting from the cardiovascular disease. Several radionuclides are available for monitoring blood flow to cardiac tissue. The most validated radionuclides for these measurements are ^{13}N , ^{15}O , $^{99\text{m}}\text{Tc}$, ^{201}Tl and ^{82}Rb . Each of ^{13}N , ^{15}O and ^{201}Tl require the presence of an on-site cyclotron, whereas, ^{82}Rb and $^{99\text{m}}\text{Tc}$ require only a generator. Rubidium (Rb) is an alkali metal ion that acts biologically like potassium and accumulates in cardiac muscle tissue. Rb has a rapid blood clearance profile which allows the use of ^{82}Rb . It also has an ultra-short physical half-life of 75 sec for non-invasive evaluation of regional cardiac blood flow. There are several advantages of ^{82}Rb over other radionuclides. Having a short half-life significantly reduces the radiation dose to the patient. In addition, ^{82}Rb is a positron emitter, which gives the full advantages of PET such as image quantification with superior sensitivity. Several reports have shown superior diagnostic performances of ^{82}Rb -PET over conventional $^{99\text{m}}\text{Tc}$ -SPECT. ^{82}Rb can be produced from a generator system by the decay of its 25.6-day half-life parent ^{82}Sr . However, the ^{82}Sr parent is difficult to prepare. In routine generator production, certain purity is required to meet the specification of the product. Since there has been no the use of ^{82}Rb radionuclide for research or medical purpose in Korea, we have plans to produce ^{82}Sr with certain purity and develop a $^{82}\text{Sr}/^{82}\text{Rb}$ generator system. These studies can also be applied to remove radioactive Sr from radioactive waste waters. Because ion exchange resin, used for purification of ^{82}Sr from impurities, is also utilized to trap radioactive Sr^{2+} ions from radioactive waste water. After Fukushima Daiichi nuclear accident, interest in the treatment of radioactive waste water has surged. As one of main fission products of nuclear reactor, ^{90}Sr has been regarded as a hazardous radionuclide with half-life of about 29 years. Therefore, the investigation on ion exchange resin is important for removal of ^{90}Sr from radioactive waste water. Here, we optimized ^{82}Sr purification method using ion exchange resin to establish the most suitable procedure.

Keywords: Ion Exchange Resin, Radioactive Strontium, Rubidium-82, Generator

Corrosion Behavior Analysis of Ni-Cr Alloy in Molten NaCl-MgCl₂ With Various Metallic Chloride at 973 K

Younghwan Jeon, Jungho Hur, and Jaeyeong Park*

Ulsan National Institute of Science and Technology (UNIST)

*jypark@unist.ac.kr

Attempts to use the molten salt system in various aspects such as MSR or energy storage systems are increasing. However, there are limitations in the molten salt-assisted technique due to the harsh corrosiveness of the molten salt, and a more detailed study on salt-induced corrosion is needed to solve this problem.

In this study, corrosion behaviors of 80Ni-20Cr alloy in various salt environments such as eutectic NaCl-MgCl₂ with NiCl₂, CrCl₂, and EuCl₃ additives were investigated. Meanwhile, the corrosion acceleration effects of 80Ni-20Cr specimens were analyzed for various ceramic materials such as SiC, Al₂O₃, SiO₂, graphite, and BN, and metallic materials such as Ni-based alloy, Fe-based alloy, and pure metals in a molten salt environment. The experiments were conducted at 973 K for up to 28 days, and after the experiment, the microstructural change of the specimen was analyzed through SEM-EDS, and salt condition was analyzed by ICP-OES.

Acknowledgements

This work was supported by a National Research Foundation of Korea (NRF) grant funded by the Ministry of Science and ICT, Republic of Korea (NRF-2022M2D2A1A02063129).

Keywords: MSR, Molten Salt, Corrosion, Ni-based Alloy

Colorimetric Analysis on the Solubility of $\text{Nd}(\text{OH})_3(\text{s})$ Under Neutral to Weakly Acidic Conditions

Ranyeong Choi and Jun-Yeop Lee*

Pusan National University (PNU)

*jylee@pusan.ac.kr

With respect to the geologic repository, intrusion of groundwater has been considered as a major factor that can transfer radionuclides to the natural environment. Moreover, the migration of radionuclides in the natural groundwater system is significantly influenced by the interaction between the radionuclides and groundwater constituents. Among various hydrogeochemical reactions, hydrolysis is one of the major reactions that can affect the aqueous solubility of radionuclides. Therefore, a precise understanding of relevant chemical thermodynamic behavior is of cardinal importance for the reliable prediction of migration/retardation behavior of radionuclides in the natural groundwater system. The objective of the present work is to investigate the solubility behavior of $\text{Nd}(\text{OH})_3(\text{s})$ to provide relevant chemical thermodynamic data of Nd(III) as a chemical analogy of major radiotoxic elements such as Am(III) and Cm(III).

All the experiments were performed with Ar gas-filled glovebox under inert atmospheric condition. The aqueous Nd(III) solution was prepared by dissolution of 0.5 g $\text{NdCl}_3 \cdot 6\text{H}_2\text{O}$ (Sigma-Aldrich) in 10 ml of deionized water. The Nd(III) solid phase was precipitated by dropwise addition of ca. 10 ml of 4 M NaOH (Sigma-Aldrich). The Nd(III) precipitate was identified to be crystalline $\text{Nd}(\text{OH})_3(\text{s})$ nanorod by using XRD and TEM. For the solubility experiment, the solid $\text{Nd}(\text{OH})_3(\text{s})$ was equilibrated at the pH range from 5.0 to 9.0 at 0.1 M NaCl condition. The total concentration of the Nd(III) was quantified by using UV/Vis absorption spectroscopy and ICP-MS after the phase separation.

In the present work, the solubility behavior of the solid $\text{Nd}(\text{OH})_3(\text{s})$ phase was investigated by using colorimetric analysis. The chemical thermodynamic data obtained in this study are expected to enhance the reliability of solubility prediction for the trivalent lanthanides and actinides.

Acknowledgements

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (No. NRF-2021M2E1A108520412).

Keywords: Hydrolysis, Solubility, UV/VIS, Nd(III)

Analysis of Iodine in Various Cement Waste Particle Size

Gi Yong Kim, Jung-Weon Choi, Jung Bo Yoo, and Tae-Hong Park*

Korea Atomic Energy Research Institute (KAERI)

*parktae@kaeri.re.kr

The massive amount of radioactive waste will be generated during decommissioning of nuclear. Among the radioactive waste from these disposal process, 50-55 million tons of concrete waste are included. For safe disposal, it is very important to accurately analyze the concentration of radionuclides, especially ^{129}I and ^{131}I , contaminated concrete. ^{129}I , a long-lived radioisotope of iodine ($t_{1/2}=1.57 \times 10^7$ y), and ^{131}I ($t_{1/2}=8.04$ d) are generated from the fission of uranium in nuclear reactors. In Korea, according to the Nuclear Safety and Security Commission (NSSC) radioactive clearance level guide, the limit for radioactive clearance level of ^{129}I is less than 0.01 (Bq/g). Iodine can be absorbed, accumulate in organisms, and exhibit low energy emission compared with cesium, and cobalt. Therefore, it is essential to accurately separate and analyze iodine radioactive waste.

In this study, we focused on the determination of iodine in simulated cement waste form containing KI for the recovery of iodine. We performed cement waste form sieved through a different particle size ($0.5 \text{ mm} < \phi < 6.35 \text{ mm}$). For the separation of iodine from solid samples with low iodine content, such as soil, sediment, and cement, for sample decomposition associated with solvent extraction using CHCl_3 for separation of iodine from the matrix. The separation of iodine in cement waste particles was therefore carried out using an acid leaching method using KI containing cement particles. We observed that cement particle size decreased at 6.35 mm to 0.5 mm with iodine yield decrease at 0.840 ± 0.011 to 0.582 ± 0.010 . Thus, in this study, the acid leaching method enables to determination Iodine in cement.

Keywords: Iodine, Cement Waste, Nuclear Decommissioning Waste, Acid Leaching

Assessment of Multi-Elements Analysis Ability Using ICP-OES

Jihyun Kim, Kyungwon Suh*, Kwang-Soon Choi, Jung Bo Yoo, and Jung-Weon Choi

Korea Atomic Energy Research Institute (KAERI)

*kyungwon0707@kaeri.re.kr

We conducted multi-elements determination of reference material certified by the Inorganic Ventures, IV-26, using iCAP 7400 ICP-OES of Thermo Fisher Scientific. And we statistically evaluated analysis results by introducing the in-house proficiency evaluation method implemented at the Ministry of Food and Drug Safety. Ca, Co, Fe, Mg, Ni, and V were selected as target elements, and extended uncertainty was estimated at a confidence level of about 95% and coverage factor $k = 2$. Five parameters incurred at manufacturing process (standard solution, calibration curve, repeated measurement and dilution factor of the test sample) were considered when determining the uncertainty. En-score can be calculated using the formula $En = (x - X) / (U_{lab}^2 + U_{ref}^2)^{1/2}$ described in KS Q ISO 13528, where x , U_{lab} , X , and U_{ref} are the test results, the uncertainty of the result, and the certified value and the uncertainty of the value. And if the absolute value $|En|$ is less than 1, it can be evaluated as a satisfied value.

As a result of ICP-OES analysis, each concentration of the elements to be measured was almost similar to the certified concentration of the reference material, and the uncertainty was slightly different. Also since evaluation on multi-elements determination had an En-score within 1, it was confirmed that the analysis results satisfied En evaluation.

Keywords: Multi-Elements Analysis, ICP-OES, En-Score

Measurement of Recovery for Validation of Pretreatment Method in Solidified Cement Ion-Exchange Resin Samples

Kwang Soon Choi*, Gi Yong Kim, Jung-Weon Choi, Jung Bo Yoo, and Tae-Hong Park

Korea Atomic Energy Research Institute (KAERI)

*nkschoi@kaeri.re.kr

We established pretreatment method of solidified cement ion-exchange resin samples generated before 2003 in nuclear power plants for measurement of non-volatile radionuclide activity. A microwave digestion system (MDS) with mixed acid ($\text{HCl-HNO}_3\text{-HF-H}_2\text{O}_2$) was used to dissolve cement and to desorb non-volatile elements such as Ce, Co, Cs, Fe, Nb, Ni, Re, Sr and U from mixed ion-exchange resin. The content of Ce, Co, Fe, Nb, Ni, Re, Sr, U and Cs after pretreatment of cement plus mixed ion-exchange resin was measured by ICP-AES and ICP-MS, respectively. As iron and strontium are also present in cement, their content after dissolving a certain amount of cement was measured by ICP-AES. All elements except Nb were quantitatively recovered. Especially since the Nb recovery was low at $72.0 \pm 2.5\%$, the MDS following addition of the mixed acid to the resin was operated once more for desorbing Nb from it. Finally the recovery of Nb was over 95%. This sample pretreatment method will be applied to solidified cement ion-exchange resin samples generated in nuclear power plants for assessment of radionuclide inventory.

Keywords: Pretreatment Method, Solidified Cement Ion-exchange Resin Sample, ICP-AES/MS, Non-volatile Radionuclide Activity, Microwave Digestion System

A Development of the Quantitative Analysis Method for Np in Solutions by Various Radiometric Methods and Absorption Spectroscopy

Junhyuck Kim*, Hye-Ryun Cho, and Jae-Il Park

Korea Atomic Energy Research Institute (KAERI)

*jhim98765@kaeri.re.kr

Neptunium (Np) is one of the daughter elements included in the decay chain of Pu. The quantitative analysis of Np isotopes is required for radioactive waste characterization, research on actinide chemistry, etc. Np-237 has a long half-life (2.144 million years), but its daughter Pa-233 has a relatively short half-life (26.975 days). For this reason, after a sufficient time elapses following the chemical preparation process of the analyte, the two nuclides are in radiation equilibrium in the sample. Np-237 emits alpha-rays while Pa-233 emits beta-rays. Both nuclides also emit gamma- and X-rays. In this study, alpha-rays were measured using liquid scintillation counting (LSC) method and alpha spectrometry. Gamma-spectrometry with a HPGe detector was used for the analysis of gamma- and X-rays. In addition, we compared the radiometric results with quantitative analysis of Np using UV-Vis absorption spectrometry. The LSC method and the HPGe gamma-spectroscopy do not require extensive sample preparation procedures. Alpha spectroscopy requires a standard material spiking, separation by coprecipitation, and disk-type sample preparation procedure to obtain measurement efficiency and recovery factor. A reference material sample with a concentration of 5.8 mM was analyzed by the four analysis methods, and all of the measured results agreed well within a difference level of 4%.

Keywords: Np-237, LSC Method, Gamma-Spectrometry, Alpha Spectrometry, UV-vis Absorption Spectrometry

Determination of Citric Acid in Radioactive Wastes Using Ion Chromatography

Hyejin Cho* and Jae Il Park

Korea Atomic Energy Research Institute (KAERI)

*hcho921@kaeri.re.kr

Complexing agents used at various nuclear facilities exist in low- and intermediate level radioactive wastes deposited in the repository site. In addition these will be generated through the degradation of the wastes such as cellulose materials. The presence of chelating agents may possibly affect the safety of the wastes repository by promoting the migration of radionuclides into geosphere. Thus, under Nuclear Safety and Security Commission's Notice No. 2021-16, the contents of chelating agents in radioactive wastes are required to be determined for the secured disposal.

UV-Vis method based on an enzymatic reaction was proved to be in adequate to analyze citric acid in radioactive wastes with complex matrix, especially for concrete. A rapid automated method using ion chromatography (IC) for analysis of citric acid in concrete samples is developed. This automated method enables a sample solution to measure without pretreatment and a target substance to separate from other concrete admixtures. Also, the developed method here, for radioactive concrete wastes was successfully applied to real samples with lowering a limit of quantification value.

Keywords: Complexing Agents, Chelating Agents, Ion Chromatography, Enzymatic Method, UV-Vis Spectrophotometer

Applications of Zone-melting to NaCl-MgCl₂ Eutectic Salt

Chan-Yong Jung and Sang-Eun Bae*

Korea Atomic Energy Research Institute (KAERI)

*sebae@kaeri.re.kr

Molten salt reactor (MSR) is one of the non-pressurized-water fourth-generation reactors that uses liquid nuclear fuel that integrates coolant and nuclear fuel, so it is a safe reactor that can fundamentally prevent severe accidents caused by coolant loss. MSR uses NaCl-MgCl₂ as a coolant salt, which is considered a promising diluent that can dissolve the fuel salt by forming an eutectic mixture. In this study, a zone-melting system was used to remove impurities from the NaCl-MgCl₂ used in MSR. The system was designed in detail to control eutectic salt impurities by traversing long charges into a small molten zone.

Keywords: MSR, NaCl-MgCl₂, Zone-melting, Eutectic Salt

Optimization of Synthetic Conditions for Titanium-containing Anderson-type Polyoxometalate

Hye Ran Noh^{1,2}, Jong-Yun Kim^{1,2}, Tae-Hyeong Kim², Jeongmook Lee^{1,2}, Jung Hwan Park^{2,3},
and Sang Ho Lim^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

³Korea Advanced Institute of Science and Technology (KAIST)

*slim@kaeri.re.kr

Anderson-type polyoxometalate (POM) with general formula of $[H_y(XO_6)M_6O_{18}]^{n-}$ ($y=0-6$, $n=2-8$, M =addenda atom, X =heteroatom) represents one of the basic topological structures among POM-type family. Anderson-type POMs have a planar arrangement and two terminal oxygen atoms attached to each addenda metal atom unlike other type. Thus, the Anderson-type POMs have high reactivity and various coordination modes. The various multifunctional organic-inorganic hybrid materials can be synthesized using the Anderson-type POMs as an inorganic building block. Another important feature of the Anderson-type POMs is the incorporation of the heteroatoms with various sizes and oxidation states, which can lead to tune chemical properties. No Anderson-type POMs with early transition metal ions in the heteroatom site have been reported previously. Recently, we reported the synthesis of titanium-containing Anderson-type POM, $Na_2K_6Ti_{0.92}W_{6.08}O_{24} \cdot 12H_2O$ (Ti-POM), which consists of pure inorganic framework built from a central Ti atom surrounded by six WO_6 inorganic scaffold.

Herein, in-depth studies were conducted to find optimal synthesis conditions such as composition and pH. The success of synthesis was confirmed with Powder X-ray Diffraction that the Ti-POM has a rhombohedral structure with space group of $R\bar{3}m$ (No. 166) when the $TiOSO_4 \cdot xH_2SO_4 \cdot yH_2O / Na_2WO_4 \cdot 2H_2O$ molar ratio is in the range of 0.07 to 0.33. But outside of this range, other unwanted phases coexist. In a basic condition (pH=12), a single-phase Ti-POM with good crystallinity can be obtained, while a Keggin-type POM, $Na_xK_{10-x}(H_2W_{12}O_{42})$, was formed through the decomposition of Ti-POM at pH lower than 7.

Keywords: Anderson-type, Polyoxometalate, Titanium, Synthesis, pH, Composition

Effect of Colloids on Safety Assessment of High-Level Waste Repository

Junghwan Park^{1,2,*} and Sang Ho Lim^{1,3}

¹Korea Atomic Energy Research Institute (KAERI)

²Korea Advanced Institute of Science and Technology (KAIST)

³University of Science and Technology (UST)

*jpark7@kaeri.re.kr

In Korea, 483,102 assemblies of spent fuel have been discharged and stored in sites, as of 2019. However, total capacity for site storage is 529,748 assemblies, and more than 90% is already saturated. Wolsong site, the most saturated site, started to construct more dry storage to extend the capacity in 2020. Spent fuel and high-level waste (HLW) is a big concern in Korean nuclear industry. Then, master plan for management of spent fuel is once announced by Ministry of Trade, Industry and Energy (MOTIE) in 2016 and reviewed by civil committee in 2019. The core contents of the plan are establishing schedule for construction of HLW management facility in one area, and construction of temporary dry storage in each site, if unavoidable. For HLW management facility, there are three following schedules: siting of Underground Research Laboratory (URL) and Interim Storage by 2020, operation of facilities initiated by 2030, and operation of final disposal facility initiated by 2050.

Final repository will be designed as deep geological repository. The concept of deep geological disposal is that spent nuclear fuel is placed in disposal containers that can withstand corrosion and pressure in long-term, permanently isolated from the human sphere of life, and dumped in deep geological media, such as crystalline rocks and clay layer, at a depth of 300 to 1,000 meters underground. The safety assessment of waste disposal sites focuses on determining whether the disposal sites meet the safety requirements of national regulatory authority. This safety assessment evaluates the potential radiation dose of radionuclides from the disposal site to humans or the environment. In this case, the calculation is performed assuming that all engineering barriers of the disposal site have collapsed in a long-term period. Then radionuclides are released from the waste, and migrated in groundwater. The dose resulting from the release and migration of radionuclides on the concentration of nuclides in groundwater. In general, metallic nuclides may exist in water in various ionic states, but some form colloids. This colloid allows more nuclides to exist in water than in solubility. Therefore, more doses may occur than we know generally predict. To determine the impact of colloids, we performed the safety assessment of the Yucca Mountain repository as an example.

Keywords: High-level Waste, Spent Fuel, Deep Geological Disposal, Colloids

High-temperature Viscosity Measurement System for Corrosive NaCl-MgCl₂ Molten Salt Using Brookfield Rotational Viscometer

Eunkyeom Lee, Chanyong Jung, Tae-Hyeong Kim, Sang Ho Lim, and Jong-Yun Kim*

Korea Atomic Energy Research Institute (KAERI)

*k jy@kaeri.re.kr

Viscosity is a fundamental physical property that is important in any system in which fluid movement occurs. In addition, most of the elements exist as ions in molten state in high-temperature molten salt, and electrical conductivity in such molten state is closely related to viscosity as a transport property. Molten salt reactor (MSR) and pyroprocess are representative processes dealing with high-temperature molten salts, actinide elements, and other radioactive materials. In MSR and pyroprocesses, the viscosity data must be provided as one of the fundamental physical property data required for safe process operations and countermeasures to severe accidents. In order to measure the viscosity of highly corrosive molten salt at high temperatures, we have built a in-house developed molten salt viscosity measurement system based on the Brookfield rotationary viscometer. We also developed a special correction technique to improve the accuracy of the viscosity measurement. In this study, the viscosity was measured at 500°C for NaCl-MgCl₂ molten salt, which is selected as the base salt material of MSR system under development in Korea Atomic Energy Research Institute (KAERI), using our viscosity measurement system installed in a oxygen- and moisture-free Ar-atmosphere glovebox. Our viscosity measurement system was calibrated using a LiCl-KCl eutectic mixture with well-known viscosity value, and viscosity values obtained using our own correction methodology were compared with those of other conventional correction methods. In our further study, we plan to measure the NaCl-MgCl₂-UCl₃ system at various compositions and temperatures.

Keywords: Molten Salt, Viscosity, Rotational Viscometer, Sodium Chloride, Magnesium Chloride

Synthesis and Characterization of New Hydrogen-Bonded Organic Framework With Subsequent Application for Iodine Capture

Viktoriya Kim^{1,2}, Jeongmook Lee^{1,2}, Tae-Hyeong Kim², Jong-Yun Kim^{1,2}, and Sang Ho Lim^{1,2,*}

¹University of Science and Technology (UST)

²Korea Atomic Energy Research Institute (KAERI)

*slim@kaeri.re.kr

Hydrogen-bonded organic frameworks (HOFs) are a new type of porous crystalline material that are constructed by intermolecular hydrogen bonding of organic building blocks to form two-dimensional (2D) and three-dimensional (3D) crystalline networks. High-quality HOF single crystals are easily grown for direct superstructure analysis using single crystal X-ray diffraction, which is essential for revealing the relationship between structure and properties. The unique advantages of HOF, such as high crystallinity, porosity and fast regeneration, have allowed it to be used in a variety of applications including catalysis and gas separation.

Squaric acid (SQA) is a non-carboxylic, organic acid with proton donor and acceptor ability which is known to take on a variety of coordination modes with metal ions. Pyrazine is a six-membered aromatic heterocycle bearing two nitrogen atoms, which has sp^2 hybridized C atoms with C-H hydrogen bonds.

This work describes the synthesis and structural characteristics of HOF based on squaric acid and pyrazine. Based on single crystal X-ray diffraction data, this MOF crystallizes in the triclinic P-1 space group. Each asymmetric unit is composed of H_2SQ and pyrazine. All squaric acid molecules share one H atom with the N atom of pyrazine molecules. The layer distance between nearby O atoms from squaric acid in different layers equals 5.29 Å. Also, our HOF showed high adsorption capacity during experiments. The composition and comparative characteristics of HOF are given using SC-XRD, PXRD, SEM and UV-vis.

Keywords: Hydrogen-bonded Organic Frameworks, Single Crystal, Porous Materials, Iodine Adsorption

2022 추계 학술발표회논문요약집

발행인 강문자

발행처 한국방사성폐기물학회

주 소 대전광역시 유성구 대덕대로 989번길 111
한국원자력연구원 원자력사업기술지원동(E7) 202호

연락처 Tel +82-42-861-5851, +82-42-866-4205

Fax +82-42-861-5852

E-mail krs@krs.or.kr



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

Abstracts of Proceedings of the
Korean Radioactive Waste Society **AUTUMN** 2022

www.krs.or.kr